No. 20-70899

**IN THE UNITED STATES COURT OF APPEALS
FOR THE NINTH CIRCUIT**

IN RE PUBLIC WATCHDOGS,

Petitioner,

v.

UNITED STATES NUCLEAR REGULATORY COMMISSION,

Respondent.

SOUTHERN CALIFORNIA EDISON COMPANY,
Intervenor.

INTERVENOR SOUTHERN CALIFORNIA EDISON'S SUPPLEMENTAL EXCERPTS OF RECORD

VOLUME 1 OF 8

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INTERVENOR SOUTHERN CALIFORNIA EDISON'S

SUPPLEMENTAL EXCERPTS OF RECORD:

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EDISON®
An EDISON INTERNATIONAL® Company

Thomas J. Palmisano
Vice President & Chief Nuclear Officer

10 CFR 50.82(a)(4)(i)

September 23, 2014

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington D.C. 20555-0001

Subject: **Docket Nos. 50-361 and 50-362, San Onofre Nuclear Generating Station, Units 2 and 3 Post-Shutdown Decommissioning Activities Report**

Reference Letter from P.T. Dietrich (SCE) to the U.S. Nuclear Regulatory Commission dated June 12, 2013; Subject: Certification of Permanent Cessation of Power Operations, San Onofre Nuclear Generating Station, Units 2 and 3

Dear Sir or Madam:

On June 12, 2013, in accordance with 10 CFR 50.82(a)(1)(i), Southern California Edison (SCE) submitted the referenced letter to the U.S. Nuclear Regulatory Commission (NRC) certifying the permanent cessation of operations at San Onofre Nuclear Generating Station (SONGS), Units 2 and 3. In accordance with 10 CFR 50.54(bb) and 10 CFR 50.82(a)(4)(i), SCE is required to submit an Irradiated Fuel Management Plan (IFMP), Site Specific Decommissioning Cost Estimate (DCE) and Post-Shutdown Decommissioning Activities Report (PSDAR) within two years of permanent cessation of operations.

The SONGS, Units 2 and 3 PSDAR is attached. The SONGS, Units 2 and 3 IFMP and DCE are being concurrently submitted under separate cover letters. The descriptions of decommissioning activities and phases in the PSDAR are consistent with those described in the DCE. Both the PSDAR and DCE represent SCE's current plans and are subject to change as the project progresses.

Changes to significant details will be included in subsequent revisions to the PSDAR as required by 10 CFR 50.54(bb). Financial assurance information will be provided on an annual basis as required by 10 CFR 50.75(f)(1).

This letter does not contain any new commitments.

If there are any questions or if additional information is needed, please contact me or Ms. Andrea Sterdis at (949) 368-9985.

Sincerely,

P.O. Box 128
San Clemente, CA 92672
(949) 368-6575 PAX 86575
Fax: (949) 368-6183
Tom.Palmisano@sce.com

A001
NRR

Enclosure: San Onofre Nuclear Generating Station Units 2 and 3 Post-Shutdown Decommissioning Activities Report

cc: M. L. Dapas, Regional Administrator, NRC Region IV
T. J. Wengert, NRC Project Manager, San Onofre Units 2 and 3 Decommissioning
R. E. Lantz, NRC Region IV, San Onofre Units 2 and 3
G. G. Warnick, NRC Senior Resident Inspector, San Onofre Units 2 and 3
S. Y. Hsu, California Department of Health Services, Radiologic Health Branch

San Onofre Nuclear Generating Station Units 2 and 3
Post-Shutdown Decommissioning Activities Report

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List of Acronyms and Abbreviations

AADT	Average Annual Daily Traffic
AIF	Atomic Industrial Forum
ALARA	As Low As Reasonably Achievable
BMP	Best Management Practices
CCC	California Coastal Commission
CFR	Code of Federal Regulations
CRWQCB	California Regional Water Quality Control Board
CSLC	California State Lands Commission
DBA	Design Basis Accident
DCE	Decommissioning Cost Estimate
Decon Pd	License Termination Period
DGC	Decommissioning General Contractor
DOE	United States Department of Energy
DOT	United States Department of Transportation
DSC	Dry Storage Canister
FES	Final Environmental Statement, SONGS Units 2 and 3 (NUREG-0490)
GEIS	Generic Environmental Impact Statement (NUREG-0586)
GTCC	Greater than Class C
HSM	Horizontal Storage Modules
IFMP	Irradiated Fuel Management Plan
ISFSI	Independent Spent Fuel Storage Installation
LTP	License Termination Plan
LLRW	Low Level Radioactive waste
MARRSIM	Multi-Agency Radiation Survey and Site Investigation Manual
MWDOC	Municipal Water District of Orange County
MWt	Megawatt-thermal
NEI	Nuclear Energy Institute
NPDES	National Pollutant Discharge Elimination System
NRC	United States Nuclear Regulatory Commission
ORISE	Oak Ridge Institute for Science and Education
PSDAR	Post-Shutdown Decommissioning Activities Report
PWR	Pressurized Water Reactor
RCS	Reactor Coolant System
REMP	Radiological Environmental Monitoring Program
RV	Reactor Vessel
SONGS	San Onofre Nuclear Generating Station
SCE	Southern California Edison
SDAPCD	San Diego Air Pollution Control District
SFP	Spent Fuel Pool
SNF Pd	Spent Fuel Period
SFSM	Spent Fuel Storage Modules
SPCC	Spill Prevention Control and Countermeasures
SR Pd	Site Restoration Period
SSC	Structures, Systems, and Components
UFSAR	Updated Final Safety analysis Report
USCB	United States Census Bureau

San Onofre Nuclear Generating Station Units 2 and 3
Post-Shutdown Decommissioning Activities Report

I. INTRODUCTION AND SUMMARY

A. Introduction

1. Historical Perspectives

San Onofre Nuclear Generating Station (SONGS) Units 2 and 3 have been owned by four entities. Two are municipalities (Riverside and Anaheim) and two are investor owned utilities: San Diego Gas & Electric (SDG&E) and Southern California Edison (SCE, the Owner-Operator and agent for the participants). The relative obligation for operation and decommissioning varies by unit and entity. The term "SONGS Participants" is used in this report to represent the four entities that have continuing decommissioning obligations.

SONGS Unit 1 was shut down in 1992 with on-shore facilities largely dismantled by 2009 and off-shore conduits being fully dispositioned this year (2014). The decision has been made to shut down and decommission Units 2 and 3. Since the decision to shut down SONGS Units 2 and 3, the focus of SONGS staff and other personnel has been to plan and begin execution of the necessary steps to achieve timely, cost-effective, and safe decommissioning and restoration of the SONGS site.

In developing its plans, SONGS has benchmarked the experiences of commercial decommissioning projects in the 1990s and 2000s and has sought the input from experienced individuals and groups with a wide range of such experience. SONGS maintains close communications with those facilities currently undergoing decommissioning and with many of the organizations supporting those efforts. In particular, both the Zion and Humboldt Bay plants are currently undergoing active decommissioning. Three others (Kewaunee, Crystal River 3, and Vermont Yankee) are, or soon will be, entering SAFSTOR conditions of varying durations prior to dismantlement.

Earlier decommissioning projects faced a number of first-time technical challenges, such as cutting reactor vessel (RV) internals in a high radiation environment. SONGS' reviews indicate that many of the technical challenges confronting SONGS decommissioning now have mature solutions. Similarly, our predecessors provide a wealth of knowledge to minimize worker radiation exposure, efficiently plan, and sequence a decommissioning project and safely manage and transport waste.

The SONGS Participants have the responsibility to restore the site in accordance with applicable regulations and agreements. The SONGS Participants have a responsibility to their stakeholders and the communities they serve to do so in a transparent and effective manner while striving to attain high standards of safety and environmental protection. Further, the SONGS Participants will have a limited, if any, role in the future use of the site. The ultimate use for the site is for the land-owner (U.S. Navy) to determine with input from the community at large.

2. Community Engagement

A key lesson-learned in our review of other decommissioning projects is the continued importance of community engagement during the decommissioning process. The SONGS Participants are committed

to engaging the local community and its leaders in an open, transparent, and proactive manner. SONGS is actively engaged with external stakeholders to: understand their priorities; inform them of SONGS plans; and, to seek their input on the safe, timely, and cost-effective decommissioning of SONGS.

The SONGS Participants are actively engaging with the community through public outreach including briefings for community groups and routine educational updates for local, state, and federal officials. The SONGS participants have formed the Community Engagement Panel (CEP) with members representing a broad range of stakeholders to advise SONGS on decommissioning matters. The panel meets at least quarterly to facilitate dialogue and includes several representatives of government, members from academia, labor, business, environmental organization, and a local anti-nuclear leader. Members of the CEP were provided with the opportunity to review and provide input on this document as well as the Decommissioning Cost Estimate (DCE) and the Irradiated Fuel Management Plan (IFMP). As a precursor to review of these submittals, SONGS hosted two workshops with external technical experts to provide the CEP members with a depth of knowledge in these areas. Feedback from the panel was addressed prior to finalization and SCE senior management authorization of the submittals.

SONGS also has established a website, www.SONGScommunity.com, as a dedicated online source for information on the plant and the decommissioning process. The website includes background information on decommissioning, links to other websites including the NRC, and an "opt-in" feature that allows members of the community to register for automatic updates on decommissioning matters.

3. Regulatory Basis

In accordance with the requirements of 10 CFR 50.82, "Termination of License," paragraph (a)(4)(i), this report constitutes the Post-Shutdown Decommissioning Activities Report (PSDAR) for SONGS Units 2 and 3. The PSDAR contains the following:

1. A description of the planned decommissioning activities along with a schedule for their accomplishment.
2. A site-specific DCE including the projected cost of managing irradiated fuel and site restoration (being submitted concurrently).
3. A discussion that provides the basis for concluding that the environmental impacts associated with the site-specific decommissioning activities will be bounded by the appropriate previously issued generic and plant specific environmental impact statements.

The PSDAR has been developed consistent with NRC Regulatory Guide 1.185, Revision 1, "Standard Format and Content for Post-Shutdown Decommissioning Activities Report." This report is based on currently available information; however, the plans discussed may be modified as additional information becomes available or as circumstances change. As required by 10 CFR 50.82(a)(7), SCE will notify the Nuclear Regulatory Commission (NRC) in writing before performing any decommissioning activity inconsistent with, or making any significant schedule change from, those actions and schedules described in the PSDAR, including changes that significantly increase the decommissioning cost.

The IFMP and DCE are being submitted concurrently with the PSDAR. The technical, schedule, and cost information provided is consistent among these submittals.

B. Background

The SONGS site is located on the coast of southern California in San Diego County, approximately 62 miles southeast of Los Angeles and 51 miles northwest of San Diego. The site is located entirely within the boundaries of the United States Marine Corps Base Camp Pendleton. The site is approximately 4,500 feet long and 800 feet wide, comprising 84 acres. The site does not include office buildings and related facilities located east of Interstate 5 (I-5) referred to as "the Mesa" or other adjacent parcels.

The property on which the station is built is subject to an easement from the United States Government through the U. S. Navy. The nearest privately owned land is approximately 2.5 miles from the site.

SONGS Units 2 and 3 is a two-unit site with supporting facilities. The reactors were previously licensed to produce 3,438 MWt each. An on-site Independent Spent Fuel Storage Installation (ISFSI) used to store SONGS Units 1, 2 and 3 fuel, located on the portion of the site previously occupied by SONGS Unit 1. Storage at the ISFSI was initiated in 2003 and the pad was subsequently (2007) expanded to support the currently placed 63 Horizontal Storage Modules in which 51 Dry Storage Containers (DSCs) have been installed to-date: 50 containing irradiated fuel and one (1) containing Greater-Than-Class-C (GTCC) materials. The most recent loading campaign was conducted in 2012. As discussed in the Spent Fuel Management Period details and the concurrently submitted IFMP, it will be necessary to further expand the current ISFSI capacity to store the complete inventory of Units 2 and 3 spent fuel. The location, capacity, and technology to be employed have not yet been finalized.

A brief history of the major milestones related to plant construction and operation is as follows:

	UNIT 2	UNIT 3
• Construction Permit Issued	October 18, 1973	October 18, 1973
• Operating License Issued	February 16, 1982	November 15, 1982
• Full Power Operation	June 15, 1983	November 18, 1983
• Final Reactor Operation	January 9, 2012	January 31, 2012

On June 7, 2013, SCE announced its decision to permanently cease power operations and decommission SONGS Units 2 and 3. By letter dated June 12, 2013 (Reference 3), SCE notified the NRC of its decision to permanently cease power operations. SCE has submitted two letters dated July 22, 2013 (Reference 5) and June 28, 2013 (Reference 4) certifying that fuel has been removed from the Unit 2 and 3 reactors, respectively.

Pursuant to 10 CFR 50.51(b), "Continuation of License," the license for a facility that has permanently ceased operations, continues in effect beyond the expiration date to authorize ownership and possession of the facility until the NRC notifies the licensee in writing that the license has been

terminated. During the period that the license remains in effect, 10 CFR 50.51 (b) requires the licensee to:

(1) Take actions necessary to decommission and decontaminate the facility and continue to maintain the facility, including, where applicable, the storage, control and maintenance of the spent fuel, in a safe condition, and

(2) Conduct activities in accordance with all other restrictions applicable to the facility in accordance with the NRC regulations and the provisions of the specific 10 CFR part 50 licenses for the facility.

C. Summary of Decommissioning Alternatives

The NRC has evaluated the environmental impacts of three general methods for decommissioning power reactor facilities in NUREG-0586, "Final Generic Environmental Impact Statement (GEIS) on Decommissioning Nuclear Facilities," Supplement 1 (Reference 6). The three general methods are:

- **DECON:** The equipment, structures, and portions of the facility and site that contain radioactive contaminants are promptly removed or decontaminated to a level that permits termination of the license after cessation of operations.
- **SAFSTOR:** The facility is placed in a safe stable condition and maintained in that state (safe storage) until it is subsequently decontaminated and dismantled to levels that permit license termination. During SAFSTOR, a facility is left intact or may be partially dismantled, but the fuel has been removed from the reactor vessel and radioactive liquids have been drained from systems and components and then processed. Radioactive decay occurs during the SAFSTOR period, thus reducing the levels of radioactivity in and on the material and potentially the quantity of radioactive material that must be disposed of during the decontamination and dismantlement.
- **ENTOMB:** Radioactive structures, systems, and components are encased in a structurally long-lived substance such as concrete. The entombed structure is appropriately maintained and continued surveillance is carried out until the radioactivity decays to a level that permits termination of the license.

The SONGS Participants have chosen the DECON method. SONGS is currently in the planning period during which the site is preparing for safe and orderly transition to dismantlement. More specifically:

- Permanent cessation of operations was announced on June 7, 2013.
- DECON methodology was selected (prompt decontamination and dismantlement after initial planning period).
- Additional ISFSI capacity will be added to meet all of the site's needs.
- Initial site characterization activities are underway.
- Plans to isolate the Spent Fuel Pools (referred to as "islanding") are in development.
- Other necessary actions to facilitate safe system retirement and removal (referred to as "cold and dark") are in development.

When the required regulatory reviews, planning, and preparation are sufficiently complete, the site will move into active decontamination and dismantlement. Current plans are for that period to overlap with completion of the relocation of spent fuel from the Spent Fuel Pools to the ISFSI.

The SONGS facility will be decontaminated and dismantled (D&D) to levels that permit termination of the NRC licenses and in accordance with the requirements agreed to by the United States Navy in the easement for the site. In support of this and in accordance with 10 CFR 50.82(a)(9), a License Termination Plan will be developed and submitted for NRC approval at least two years prior to termination of the license.

The decommissioning approach for SONGS is described in more detail in the following sections:

- Section II summarizes the planned decommissioning activities and general timing of their implementation.
- Section III summarizes the cost estimating methodology employed by Energy*Solutions* and references the site specific DCE being submitted concurrently.
- Section IV describes the basis for concluding that the environmental impacts associated with decommissioning SONGS Units 2 and 3 are bounded by the most recent site-specific environmental impact statement and NRC GEIS related to decommissioning.

II. DESCRIPTION OF PLANNED DECOMMISSIONING ACTIVITIES

The SONGS Units 2 and 3 decommissioning project is currently in the planning period transitioning to DECON as soon as necessary planning, approvals, and conditions permit doing so in a safe and cost-effective manner. DECON is defined in Section I.C of this report.

Table II-1 provides a summary of the current decommissioning plan and schedule for SONGS Units 2 and 3. The major decommissioning periods and general sequencing of the activities that will occur during each period identified in Table II-1 are discussed in more detail in the sections that follow. The periods are logical groupings of activities. The categories are also consistent with the Nuclear Decommissioning Trust (NDT) funds which are allocated based on specific regulatory requirements. The activities executed during these periods will, in many cases progress in parallel, and may not be as completely segregated as the description implies. For instance, while distinct decontamination and dismantlement activities are listed, it may be determined to be more effective from dose, labor, or waste disposal perspectives to dismantle structures and systems and dispose of them as radioactive waste rather than decontaminate them and dispose of the balance as non-radioactive waste.

The planning required for each decommissioning activity, including the selection of the process to perform the work, will be performed in accordance with appropriate governance and oversight processes. Based on current plans, no decommissioning activities unique to the site have been identified and no activities or environmental impacts outside the bounds considered in the GEIS have been identified. Appropriate radiological and environmental programs will be maintained throughout

the decommissioning process to ensure radiological safety of the workforce and the public and environmental compliance is maintained.

Table II-1
San Onofre Nuclear Generating Station Units 2 and 3
Current Schedule of Decommissioning Periods

Task Name	Start	Finish
Part 50 License Termination (other than ISFSI)		
Announcement of Cessation of Operations	06/07/2013	N/A
Decon Period 1 – Transition to Decommissioning	06/07/2013	12/31/2013
Decon Period 2 – Decommissioning Planning and Site Modifications	01/01/2014	06/30/2015
Decon Period 3 – Decommissioning Preps/Reactor Internals Segmentation	06/30/2015	06/01/2019
Decon Period 4 – Plant Systems and Large Component Removal	06/01/2019	09/24/2022
Decon Period 5 – Building Decontamination	09/24/2022	07/13/2024
Decon Period 6 – License Termination During Demolition	07/13/2024	12/24/2032
Spent Fuel Management		
SNF Period 1 – Spent Fuel Management Transition	06/07/2013	12/31/2013
SNF Period 2 - Spent Fuel Transfer to Dry Storage	01/01/2014	06/01/2019
SNF Period 3 – Dry Storage During Decommissioning – Units 1, 2 & 3	06/01/2019	12/05/2031
SNF Period 4 – Dry Storage Only – Units 1, 2 & 3	12/05/2031	12/31/2035
SNF Period 5 – Dry Storage Only – Units 2 & 3	12/31/2035	12/31/2049
SNF D&D Period 1 – ISFSI Part 50 License Termination	12/31/2049	05/06/2050
SNF D&D Period 2 – ISFSI Demolition	05/06/2050	09/08/2051
Site Restoration		
SR Period 1 – Transition to Site Restoration	06/07/2013	06/30/2015
SR Period 2 – Building Demolition During Decommissioning	06/30/2015	07/11/2017
SR Period 3 – Subsurface Demolition Engineering and Permitting	10/01/2019	07/13/2024
SR Period 4 – Building Demolition to 3 Feet Below Grade	07/13/2024	10/14/2028
SR Period 5 – Subgrade Structure Removal Below -3 Feet	10/14/2028	12/5/2031
SR Period 6 – Final Site Restoration and Easement Termination	05/06/2050	12/15/2051
Final Easement Termination	12/15/2051	N/A

Note [1]: Shipping dates are assumed based on the previously documented positions of the DOE, which indicates that shipments from the industry could begin as early as 2024 and SONGS place in the current queue. Both are subject to changes.

A. Detailed Breakdown of License Termination Periods

The License Termination Periods (referred to as decontamination periods) include those activities necessary to remove or reduce the levels of radioactive contamination to levels necessary to terminate the Part 50 licenses for the site (other than the ISFSI) and release it back to the Navy. Also included are the development, submittal, and support for the review of the primary decommissioning documents.

Periods 1 and 2 generally consist of planning and transition of the site to a condition where it is ready for significant decontamination and dismantlement activities. As detailed below, these periods include: system abandonment and isolation of the remaining structures, systems and components (SSC) from normal power and water sources. System abandonment and isolation allow the decontamination and dismantlement to proceed safely and in an efficient sequence. Additionally, the selection of the contractor for managing the bulk of the decommissioning activities will be made.

Period 3 is focused on decontamination and dismantlement of the major components in the containment building (RV internals, vessel, head, steam generators, pressurizer, and main piping).

Period 4 addresses the decontamination and dismantlement of SSCs known to be substantially contaminated and the removal of the components from both Periods 3 and 4.

Period 5 is focused on decontamination of the various buildings. As noted elsewhere it may be more appropriate to simply proceed with dismantlement if it is more timely and cost-effective to simply dispose of building material as radioactive waste.

Period 6 is focused on the final site survey to confirm that the site is acceptable for release back to the Navy. The process for doing so "Multi-Agency Radiation Survey and Site Investigation Manual" (MARRSIM) was developed by the four federal agencies having authority over radioactive materials (Department of Defense, Department of Energy, the Environmental Protection Agency and the NRC) and is the consensus standard endorsed by other stakeholders. Its application will be validated by the NRC.

Decontamination Period 1 – Transition to Decommissioning

- Announcement of Cessation of Operations
- Defuel Reactors
- Notification of Permanent Fuel Removal
- Disposition of legacy Low Level Radioactive Waste (LLRW)

Decontamination Period 2 – Decommissioning Planning and Site Modifications

- Preparation of Decommissioning Related Licensing Submittals
    - Permanently Defueled Technical Specifications (Submitted March 21, 2014)
    - Permanently Defueled Radiological Emergency Plan (Submitted March 31, 2014)
- Submit PSDAR, DCE and IFMP to NRC
- Perform Historical Site Assessment and Site Characterization
- Planning, Design, and Implementation of Cold and Dark (Site Repowering)
- Design and Install Spent Fuel Pool Islanding, Control Room Relocation, and Security Modifications
- Select Decommissioning General Contractor (DGC)

Decontamination Period 3 – Decommissioning Preparations and Reactor Internal Segmentation

- DGC Mobilization and Planning
- System Decontamination
- Reactor Internals Removal Preparations

- Reactor Internals Segmentation Planning and Implementation
- Purchase Dry Storage Canisters for GTCC Waste
- Segment and Package Reactor Internals for Storage in the ISFSI

Decontamination Period 4 – Plant Systems and Large Component Removal
- Upgrade Rail Spur in Owner Controlled Area
- Install Large Array Radiation Detection System to Monitor Shipments In/Out of Site
- Remove, Package, and Dispose of Non-Essential Systems
- Asbestos and Lead Abatement
- Spent Fuel Pool Closure
- Remove Spent Fuel Pool Racks, Spent Fuel Pool Island Equipment, and Bridge Crane
- Remove and Dispose of Legacy Class B and C Wastes
- Remove, Package, and Dispose of Essential Systems
- Removal and Disposal of Spent Resins, Filter Media, and Tank Sludge
- Large Component Removal
- Prepare License Termination Plan

Decontamination Period 5 – Building Decontamination
- Decontaminate Containment Buildings
- Decontaminate Turbine Buildings
- Decontaminate Fuel Handling Buildings
- Decontaminate Auxiliary Rad-waste Building
- Decontaminate Auxiliary Control Building
- Decontaminate Penetration Buildings
- Decontaminate Safety Equipment and Main Steam Isolation Valve (MSIV) Buildings
- Radiological Survey of Structures During Decontamination

Decontamination Period 6 – License Termination
- Final Status Survey
- Verification and NRC Approval

B. Detailed Breakdown of Spent Fuel Management Periods

The Spent Nuclear Fuel Management Periods began with all spent fuel off-loaded from the reactor vessel into the Spent Fuel Pools and the certification of permanent defueling letters submitted to the NRC in accordance with 10 CFR 50.82(a)(1)(ii) (References 4 and 5).

During Period 1 measures will be planned, designed, and implemented to ensure spent fuel storage and handling systems will continue to function to support fuel storage in the spent fuel pool and to facilitate transfer of the spent fuel to the ISFSI. Systems, structures, and programs needed to support the safe storage and transfer of spent fuel such as security, fire protection, and environmental and radiological monitoring will be maintained in accordance with applicable requirements. Equipment maintenance, inspection, and operations will be performed on these systems and structures as appropriate.

During Period 2 the ISFSI capacity will be expanded to accommodate transfer of all spent fuel to dry storage. All spent fuel for Units 1, 2 and 3 will be transferred to the ISFSI and stored there until it is accepted by the Department of Energy (DOE) and transferred to an off-site facility.

The next three periods reflect slightly different ISFSI conditions. Period 3 is concurrent with ongoing site decontamination and dismantlement activities. Period 4 reflects the ISFSI with spent fuel from all three units in dry storage and Period 5 recognizes the potential that Unit 1 fuel may be accepted by the DOE earlier than Units 2 and 3 fuel and ends with DOE acceptance of all Units 2 and 3 fuel.

The SNF D&D Periods (1 and 2) follow DOE acceptance and may be well after License Termination for the balance of the site.

Spent Nuclear Fuel Period 1 – Spent Fuel Transfer Management Transition
- Implementation of Initial Security Enhancements Required for Reductions in Staff
- Design and Fabricate Dry Storage Canisters for Current ISFSI Scope

Spent Nuclear Fuel Period 2 – Spent Fuel Transfer to Dry Storage
- Submit IFMP
- Select Dry Storage System Canister Design and Vendor for Balance of the ISFSI
- Design and Construct ISFSI Expansion
- Purchase, Deliver, and Load Dry Storage Canisters and Storage Models for Balance of the ISFSI
- Complete Transfer of Spent Fuel to ISFSI

Spent Nuclear Fuel Period 3 – Dry Storage during Decommissioning Units 1, 2, and 3 Fuel

Spent Nuclear Fuel Period 4 – Dry Storage Only – Units 1, 2, and 3 Fuel

Spent Nuclear Fuel Period 5 – Dry Storage Only – Units 2 and 3 Fuel

Spent Nuclear Fuel Period D&D 1 – ISFSI License Termination
- Preparation and NRC Review of ISFSI Portion/Revision of License Termination Plan

Spent Nuclear Fuel Period D&D 2 – ISFSI Demolition

- Decontamination of Storage Modules (SFSMs)
- Final Status Survey of ISFSI
- Clean Demolition of HSM's and ISFSI Pad
- Clean Demolition of ISFSI Support Structures
- Restore ISFSI Site
- Preparation of Final Report on ISFSI Decommissioning and NRC Review

C. Detailed Breakdown of Site Restoration Periods

The Site Restoration periods reflect the planning and implementation of dismantlement activities not associated with radioactive materials. The DCE and descriptions below conservatively include activities

from which the SONGS Participants will plan to seek alternatives. These include the complete removal of the intake and discharge conduits in the Pacific Ocean currently required by the California State Lands Commission (CSLC) easement. Previously, the CSLC and SONGS developed an alternative for the SONGS Unit 1 conduits. Another is associated with removal of all subsurface structures that may be required by the US Navy easement. The typical practice has been to remove structures to that depth necessary to remove contaminated materials.

Also included as part of site restoration are severance costs and cost associated with returning the Mesa and other parcels to the U. S. Navy.

Site Restoration Period 1 –Transition to Site Restoration
- Severance Costs Associated with Staffing Reduction in Accordance with State Law
- Other off-site activities are included in the DCE but are not considered part of the Units 2 and 3 PSDAR activities

Site Restoration Period 2 –Building Demolition During Decommissioning
- Demolish South Access for Decommissioning, South Yard Facility
- Other off-site activities are included in the DCE but are not considered part of the Units 2 and 3 PSDAR activities

Site Restoration Period 3 – Subsurface Demolition Engineering and Permitting
- Hydro-geologic Investigation and Outfall Conduit Survey
- Subsurface Structure Removal Analyses for Lease Termination Activities
- Final Site Grading and Shoreline Protection Engineering Planning and Design

Site Restoration Period 4 – Building Demolition to Three Feet Below-Grade
- Demolition Preparations
- De-tension and Remove Containment Building Tendons
- Demolish Diesel Generator Buildings
- Demolish Condensate Buildings and Transformer Pads
- Demolish Full Flow Areas and Turbine Buildings
- Demolish Auxiliary Rad-waste Building
- Demolish Auxiliary Control Building
- Remove Systems and Demolish Make-up Demineralizer Structures
- Demolish Penetration Buildings
- Demolish Safety Equipment and MSIV Buildings
- Demolish Fuel Handling Buildings
- Demolish Containment Buildings
- Demolish Intake and Discharge Structures

Site Restoration Period 5 – Subgrade Structure Removal below Three Feet (if required)
- Install Sheet Piling and Excavation Shoring, Dewatering System, and Effluent Treatment and Discharge Controls

- Demolish and Backfill Subsurface Structures
- Demolish and Backfill Intake Structure Inside Seawall
- Remove Off-shore Intake and Outfall Conduits
- Remove Sheet Piling and Excavation Shoring, and Perform Dewatering and Effluent Treatment
- Finish Grading and Re-vegetate Site As Needed/Required

Site Restoration Period 6 – Final Site Restoration and Easement Termination [details subject to final resolution of negotiations with the U. S. Navy]

- Install Dewatering System and Effluent Treatment and Discharge Controls
- Remove and Stockpile Existing Seawall Erosion Protection
- Remove Seawall and Pedestrian Walkway
- Remove Remaining Intake Structure Beneath Seawall
- Backfill and Compaction of Excavation
- Remove Dewatering System and Effluent Treatment
- Remove Railroad Tracks, Stabilized Slopes, Access Road, and North Parking Lot
- Finish Grading and Re-vegetate Site as Needed/Required

D. General Decommissioning Considerations

1. Major Decommissioning Activities

As defined in 10 CFR 50.2, "Definitions," a "major decommissioning activity" is "any activity that results in permanent removal of major radioactive components, permanently modifies the structure of the containment, or results in dismantling components for shipment containing greater than Class C waste in accordance with 10 CFR 61.55." The following discussion provides a general summary of the major decommissioning activities currently planned for SONGS Units 2 and 3. These activities may be modified as conditions dictate.

Prior to starting a major decommissioning activity, the plant components will be radiologically surveyed and decontaminated, as required, to minimize worker radiation exposure. Shipping casks and other equipment necessary to conduct decommissioning activities will be designed and procured.

The initial major decommissioning activities will focus on removal, packaging and disposal of piping and components. Following RV and cavity reflood and RV head removal and disposal; the reactor vessel internals will be removed from the reactor vessel and segmented as necessary to separate the GTCC waste which will be placed in storage canisters and modules on the ISFSI designated for that purpose. Using this approach, the internals will be packaged and disposed of independent of the reactor vessel (RV). When the internals segmentation effort is completed, the RV and cavity will be drained and any remaining debris will be removed.

Removal of the reactor vessel follows the removal of the reactor internals. It is likely that the components will be removed by sectioning or segmenting performed remotely. These activities may be

performed in air, rather than underwater, using a control envelope to preclude the spread of contaminated materials.

Additional major decommissioning activities that will be conducted include removal and disposal of the steam generators, pressurizer, spent fuel storage racks, and spent fuel bridge crane. The dismantling of the containment structure will be undertaken as part of the reactor building demolition. As detailed in Section 3 (below) appropriate radiation protection and contamination control measures will be employed to manage these activities.

2. Other Decommissioning Activities

In addition to the major decommissioning activities discussed above, plant components will be removed from the Turbine Building including the turbine generator, condenser, feedwater heaters, moisture separator/reheaters, and miscellaneous system and support equipment. As detailed in Section 3 (below) appropriate radiation protection and contamination control measures will be employed to manage these activities.

3. Decontamination and Dismantlement Activities

The objectives of the decontamination effort are two-fold. The first objective is to reduce radiation levels throughout the facility to minimize personnel radiation exposure during dismantlement. The second objective is to clean as much material as possible to 'unrestricted use' levels, thereby allowing non-radiological demolition and disposal and minimizing the quantities of material that must be disposed of by costly burial as radioactive waste. The second objective will be achieved by decontaminating structural components including steel framing and concrete surfaces. The methods to accomplish this are typically mechanical, requiring the removal of the surface or surface coating and are used regularly in industrial and contaminated sites.

The decontamination and/or dismantlement of contaminated SSCs may be accomplished by: decontamination in place; decontamination and dismantlement; or dismantlement and disposal. A combination of these methods may be utilized to reduce contamination levels, worker radiation exposures, and project costs. Material below the applicable radiological limits may be released for unrestricted disposition (e.g., scrap, recycle, or general disposal). Radioactive contaminated or activated materials will be removed from the site as necessary to allow the site to be released for unrestricted use.

LLRW will be processed in accordance with plant procedures and existing commercial options. Contaminated material will be characterized and segregated for additional onsite decontamination or processing, off-site processing (e.g., disassembly, chemical cleaning, volume reduction, waste treatment), and/or packaged for controlled disposal at a low-level waste disposal facility.

Contaminated concrete and structural steel components will be decontaminated and removed as required to gain access to plant SSCs. After the SSCs are removed and processed as described above,

the remaining contaminated concrete and structural steel components will be decontaminated and/or removed. Contaminated concrete will be packaged and shipped to a low-level waste disposal facility. Contaminated structural steel components may be removed to a processing area for decontamination, volume reduction, and packaging for shipment to processing facility or to a low-level waste disposal facility, as necessary.

Buried and embedded contaminated components (e.g., piping, drains) will be decontaminated in place, or excavated and decontaminated. Appropriate contamination controls will be employed to minimize the spread of contamination and to protect personnel.

4. Radioactive Waste Management

A major component of the total cost of decommissioning SONGS Units 2 and 3 is the cost of safely packaging and disposing of contaminated SSCs, contaminated soil, resins, water, and other plant process liquids. A waste management plan will be developed consistent with regulatory requirements for each waste type. Currently, LLRW Classes B and C may be disposed of at the Waste Control Services (WCS) waste disposal site in Andrews County, Texas. The waste management plan will be based on the evaluation of available methods and strategies for processing, packaging, and transporting radioactive waste in conjunction with the available disposal facility and associated waste acceptance criteria.

Class A LLRW will be disposed at a licensed disposal site. (SONGS has contracted with Energy*Solutions* to use the facility located in Clive, Utah as well as WCS). If other licensed Class B and C LLRW facilities become available in the future, SONGS may choose to use them as well.

5. Removal of Mixed Wastes

Mixed wastes (hazardous and radioactive) generated during decommissioning, if any, will be managed in accordance with applicable Federal and State regulations. If technology, resources, and approved processes are available, the processes will be evaluated to render the mixed waste non-hazardous. Otherwise, mixed wastes from SONGS will be transported by authorized and licensed transporters and shipped to authorized and licensed facilities.

6. Site Characterization

During the decommissioning process, a site characterization will be performed in which radiological, regulated, and hazardous wastes will be identified, categorized, and quantified. Surveys will be conducted to establish the contamination and radiation levels throughout the plant. The information will be used in developing procedures to ensure the contaminated areas are removed and ensure that worker exposure is controlled. Surveys of the selected outdoor areas will also be performed including surveys of soil and groundwater near the site. As decontamination and dismantlement work proceeds, surveys will be conducted to maintain the site characterization current and ensure that decommissioning activities are adjusted accordingly.

7. Groundwater Protection

A groundwater protection program was initiated at SONGS in accordance with NEI 07-07, "Industry Groundwater Protection Initiative, Final Guidance Document," in August 2007 (Reference 11). A site hydrology study was initially completed as part of this initiative and was updated in 2012. Monitoring wells were installed around the plant to monitor for radionuclides. Acceptable levels of contaminants, as defined by the program, have been observed throughout the sampling program implemented as part of this initiative. Appropriate program elements will be maintained during decommissioning.

8. Change to Management and Staffing

With the plant shut down and defueled, plant management and staffing levels have been and continue to be adjusted to reflect the transition from an operating plant to a plant in decommissioning status. Staffing plans are addressed in the DCE.

III. ESTIMATE OF EXPECTED DECOMMISSIONING AND SPENT FUEL MANAGEMENT COSTS

10 CFR 50.82(a)(8)(iii) requires that a site-specific decommissioning cost estimate be prepared, and submitted within two years following permanent cessation of operations. 10 CFR 50.82 (a)(4)(i) requires that the PSDAR contain a site-specific decommissioning cost estimate including the projected costs of managing irradiated fuel.

Energy*Solutions* has prepared a site-specific DCE for SONGS, which also provides projected costs of managing irradiated fuel, as well as non-radiological decommissioning and other site restoration costs,. The site-specific decommissioning cost analysis is being submitted concurrent with the IFMP and this PSDAR and fulfills the requirements of 10 CFR 50.82(a)(4)(i) and 10 CFR 50.82(a)(8)(iii). A summary of the annual costs associated with decommissioning, irradiated fuel management and site restoration are provided in the Irradiated Fuel Management Plan also being concurrently submitted in accordance with 10 CFR 50.54(bb).

The methodology used by Energy*Solutions* to develop the site-specific decommissioning cost analysis follows the approach originally developed by the Atomic Industrial Forum (now Nuclear Energy Institute) in their program to develop a standardized model for decommissioning cost estimates. The results of this program were published as AIF/NESP-036, "A Guideline for Producing Commercial Nuclear Power Plant Decommissioning Cost Estimates," (Reference 7). This document includes a unit cost factor method for estimating direct activity costs, simplifying the estimating process. The unit cost factors used in the study reflect the latest available data at the time of the study concerning worker productivity during decommissioning.

The decommissioning of the SONGS site will be funded from Nuclear Decommissioning Trusts established by each SONGS Participant for each unit. The relative liabilities of each SONGS Participant are detailed in the DCE. Sufficient funds (based on balances and earnings) are projected to be available to complete the planned decommissioning activities.

As discussed in Section IV of the IFMP the CPUC will establish processes for oversight of withdrawals from the nuclear decommissioning trusts by SCE and SDG&E, and designate the specific amounts from the existing fund balances that are available for the three decommissioning cost categories: (1) spent fuel management; (2) site restoration; and (3) license termination. As entities not subject to CPUC jurisdiction, Anaheim and Riverside are not required to obtain CPUC authorization with respect to withdrawals from their respective Nuclear Decommissioning Trusts.

IV. ENVIRONMENTAL IMPACTS

As shown in this section, SCE has evaluated the environmental impacts of decommissioning SONGS Units 2 and 3 to determine if anticipated impacts are bounded by existing environmental impact statements, the NRC's generic decommissioning EIS (GEIS, Reference 6) and the SONGS Final Environmental Statement (FES, Reference 8). As noted in Regulatory Guide 1.185, C.4 "the PSDAR does not need to include the analysis of the specific environmental impacts associated with decommissioning activities....the licensee must ensure that supporting documentation and analyses are available at the reactor site for inspection by the NRC Staff." Such detailed documentation and analyses are contained in the Environmental Impact Evaluation (EIE) and its supporting references as noted in the Developmental References. They are available on-site for NRC review as well as on the SONGScommunity.com website and are summarized below. Both the detailed documentation and analyses and the following summary were reviewed by internal and external subject matter experts, independent third-party reviewers and the Community Engagement Panel discussed in the Introduction to this report.

In the GEIS, the NRC reviewed the environmental impacts resulting from decommissioning on a generic basis, and identified a need for site-specific analyses for: (1) threatened and endangered species and (2) environmental justice. In addition, site-specific analyses are called for whenever decommissioning plans indicate that activities will impact areas beyond the operational portions of a facility. The SONGS FES addresses decommissioning, but does not establish bounding environmental impacts specific to decommissioning. However, the FES' discussion of impacts for construction does describe bounding impacts as it related to potential dewatering during decommissioning.

The NRC, in its GEIS, identified additional activities that are performed in conjunction with decommissioning. These activities are regulated by the NRC but any associated environmental impacts are addressed directly in conjunction with those regulated activities. These activities include those related to the decision to permanently cease operations, irradiated fuel management in wet or dry storage, irradiated fuel transport and disposal, and the treatment, and/or disposal of LLRW. SCE similarly excluded consideration of such activities to remain consistent with the NRC's approach.

A. Environmental Impacts of Decommissioning SONGS

SCE assessed the potential for environmental impacts to each resource area from decommissioning activities using the evaluations in the GEIS as a guide. Like the GEIS, the analysis assumed that operational mitigation measures will be continued and did not rely on the implementation of new

mitigation measures unless specified. Releases to the environment, waste volumes, and other environmental interfaces were estimated in the DCE or other sources referenced in the EIE. This information was then assessed against the potential for impact and the existing environmental conditions at SONGS to identify impacts and determine whether the GEIS and FES remain bounding. The GEIS categorizes significance levels as SMALL (impacts are not detectable or are so minor that they will neither destabilize nor noticeably alter any important attribute of the resource or do not exceed permissible levels in the NRC's regulations), MODERATE (impacts are sufficient to alter noticeably, but not to destabilize, important attributes of the resource), or LARGE (impacts are clearly noticeable, and are sufficient to destabilize important attributes of the resource).

To support the evaluation, SCE established the baseline environmental and societal conditions through site-specific information as well as vicinity and regional data available from local, state, and federal agencies. In addition, the evaluation considered the existing permit conditions and limitations for water and air permits and NRC regulatory requirements, including those focused on occupational dose, public dose, radiological effluents, and LLRW shipping. Federal, state, and local requirements for non-radiological interfaces with the environment were considered. These include regulatory limits on water withdrawal and discharges, air emissions including fugitive dust, noise levels, and protection of avian, terrestrial and aquatic species, protection of cultural resources, disposal of non-radiological waste, and worker health protection.

SCE reviewed the planned decommissioning activities for SONGS Units 2 and 3 and compared these to the decommissioning activities that NRC evaluated in the GEIS. The planned activities fall within the activities that NRC evaluated. While each decommissioning site is unique, no unusual site-specific features or aspects of the planned SONGS Units 2 and 3 decommissioning have been identified. Furthermore, the practices used to accomplish the individual decommissioning tasks will employ conventional methods.

SCE's review confirmed that the anticipated or potential impacts are within the bounds of the generic impacts that the NRC described in the GEIS. There are no applicable bounding impacts for threatened and endangered species and environmental justice. The site-specific analyses determined that the planned SONGS Units 2 and 3 decommissioning activities are not likely to result in significant impacts to threatened and endangered species nor have disproportionate impacts on minority or low-income populations. The following discussions summarize the full Environmental Impact Evaluation focusing on the reasons for reaching this conclusion.

1. Onsite/Offsite Land Use

SCE's decommissioning plans include building demolition and removal within the 84-acre easement hosting the SONGS Units 2 and 3 reactor units and infrastructure. SCE plans to seek an easement lease amendment from the CSLC for the partial removal or abandonment in-place of the SONGS Units 2 and 3 intake and discharge conduits. In addition, the existing rail spur serving the site will most likely be used in support of waste shipments.

The SONGS site is currently used for utility-related industrial land uses, with the majority of the property within the easement having been previously disturbed during construction and operation of the plant. The coastal bluff areas located in the northwest and southeast portions of the 84-acre easement have remained undeveloped in compliance with the California Coastal Commission (CCC) Guarantee Agreement, in which SCE provided assurance that they will be protected and that they will remain in their natural state. It is anticipated that there will be no changes in onsite land use patterns during decommissioning.

The GEIS assessment for land use concluded that the impact would be SMALL for sites that did not require additional land for decommissioning activities. If additional land was needed the impact should be determined on a site-specific basis. Because no additional lands are needed SONGS onsite land use impacts during decommissioning are bounded by the GEIS and are categorized as SMALL.

2. Water Use

SONGS Units 2 and 3 acquires potable water through the South Coast Water District, a member agency of the Municipal Water District of Orange County (MWDOC). The site historically used water from the Pacific Ocean for its condenser cooling and service water cooling functions. The operational demand for cooling and makeup water has been significantly reduced since SONGS Units 2 and 3 permanently ceased operation. Condenser cooling is not required when the plant is not operating and service water cooling demands have been reduced to the extent possible (primarily spent fuel pool cooling). The normal operation demand was previously over 830,000 gpm per unit and is currently approximately 34,000 gpm total for both Units 2 and 3. During the decommissioning period, SONGS intends to continue to reduce cooling water demands with the intent to eliminate such demands on the Pacific Ocean as soon as possible.

The GEIS assessment of water use concluded the impact on water use would be SMALL if the decommissioning did not significantly increase water use. Water uses for decommissioning include staff usage, fuel storage (replacement of evaporative losses, etc.), fuel transfer (washing down transport casks), large component segmentation generally performed underwater, decontamination and dismantlement (if water-jet or similar techniques are employed). Water uses are anticipated to be significantly less than during operation. Thus water use impacts during decommissioning are bounded by the GEIS.

3. Water Quality – Non-Radiological

Major activities that could impact surface and groundwater quality during decommissioning include site excavation, stabilization, decontamination, dismantlement, and dewatering. These activities present the potential of spills, migration of low concentrations of radioactivity or hazardous substances not previously identified, and leaching from subsurface structures.

As discussed in Section 2 above, the site uses water from the Pacific Ocean for its condenser cooling and service water cooling functions. Water used for cooling functions is discharged through the ocean outfalls for Units 2 and 3, and is currently regulated under individual National Pollutant Discharge Elimination System (NPDES) Permits from the San Diego Regional Water Quality Control Board (SDRWQCB). The individual unit permits may be merged into a single NPDES Permit which would also continue to address groundwater dewatering discharges, and multiple minor waste stream discharges from within SONGS Units 2 and 3.

Storm water discharge is regulated and controlled through an industrial storm water general permit issued by the SDRWQCB. This permit requires SONGS to develop, maintain, and implement a storm water pollution prevention plan (SWPPP) for the facility. Storm water-related monitoring plans and reporting protocols will be updated as necessary to address permit requirements and decommissioning activities.

A previous SCE study concluded that no drinking water pathway exists for exposure from SONGS operations. Furthermore, the nearest drinking water well is more than one mile inland. Previous studies indicate that even under extreme pumping conditions, a seaward gradient will exist. Therefore, any dewatering is not expected to result in saltwater intrusion.

The GEIS assessment of water quality impacts concluded the impacts would be SMALL based on compliance with regulatory requirements including the appropriate application of best management practices (BMPs) and controls. SCE will follow standard storm water BMPs as documented in the current Industrial SWPPP and implement the current SPCC plan to minimize the chance of both groundwater and surface water contamination. In the event an unknown area of hazardous substances is identified during sub-grade soil excavation and structures removal, the area will be assessed and controlled. Due to the implementation of BMPs and compliance with permits, the potential impacts of decommissioning on nonradioactive aspects of water quality for both surface water and groundwater are bounded by those addressed in the GEIS.

4. Air Quality

Emission sources in San Diego County are primarily mobile sources (vehicular traffic) and ambient air quality standards are frequently exceeded for ozone and particulate matter due to routine vehicular traffic. Relatively minor stationary sources, such as those planned for use at SONGS, are projected to be a fraction of the average daily emissions permitted by the San Diego Air Pollution Control District (SDAPCD).

The most likely impact of decommissioning on air quality will be due to dust. SCE will employ standard dust control measures during decommissioning in accordance with SDAPCD dust abatement and visible emissions requirements. Air emissions due to commuting workers will actually be less since the work force during all phases of decommissioning is expected to be smaller than the peak number of workers used for construction or refueling outages.

The NRC's GEIS generically determined air quality impacts associated with decommissioning to be SMALL due to the sufficiency of current and commonly used control and mitigation measures. SCE will implement standard mitigation measures to reduce emissions during decommissioning per the requirements of the SDAPCD. Therefore, air quality impacts related to decommissioning of SONGS Units 2 and 3 are bounded by the GEIS.

5. Aquatic Ecology

SCE has characterized the aquatic environment in the vicinity of the SONGS Units 2 and 3 intake and discharge conduits prior to construction of and during the operation of SONGS. There are a variety of habitat types surrounding the SONGS Units 2 and 3 conduits. The marine habitat offshore of SONGS consists of a mixture of sand, cobble, and isolated areas of exposed rock. The area of high marine productivity in the immediate vicinity of the plant site is the shallow sub-tidal zone, approximately 1,300 feet north of SONGS. This area supports a biological community dominated by surfgrass, and feather boa kelp. The San Onofre kelp bed is approximately 650 feet south of SONGS Unit 2 diffusers in a water depth of 40 to 50 feet. The benthic fish community is generally dominated by queenfish; northern anchovy; white croaker and speckled sanddab.

Since ceasing permanent operations at SONGS Units 2 and 3, SCE has reduced ocean water withdrawals and discharge by approximately 96 percent from normal operating flows. The remaining flow is primarily associated with cooling spent fuel while in wet storage. As noted earlier, spent fuel storage and cooling are existing operational activities and is not re-addressed as part of this environmental review. SONGS will continue to comply with its applicable regulatory and permit requirements associated with reduction of impingement and entrainment impacts due to water withdrawals.

SCE sought and obtained an amendment to the CSLC easement lease for Unit 1 which allowed the intake and discharge conduits to remain buried beneath the seafloor. SCE is planning to pursue similar amendments for SONGS Units 2 and 3. If the CSLC approves the amendment to allow SCE to abandon the conduits in place, the environmental impacts are projected to be SMALL with the application of appropriate mitigation measures enumerated in the lease amendment. Complete removal of the conduits, as is currently required by the CSLC lease, is anticipated to have significant adverse environmental impacts. The detailed Environmental Impact Evaluation assumes the CSLC lease is amended. If the CSLC lease is not amended, the environmental impacts from complete removal of the conduits will have to be further addressed. If necessary, SCE will update the PSDAR and initiate other regulatory interactions to address the results of this analysis.

There are no surface water bodies on the SONGS site, but the Pacific Ocean borders the site and vernal pools are found northwest of SONGS Parking Lot 4. Decommissioning activities for SONGS Units 2 and 3 will include the application of common BMPs, compliance with the SONGS storm water permit, and implementation of the storm water pollution prevention plan, which will be updated as necessary to address decommissioning activities. These measures will ensure that any changes in surface water quality will be non-detectable and non-destabilizing.

The NRC determined aquatic ecology impacts to be SMALL when only aquatic resources within a plant's operational areas are disturbed. The potential impacts to aquatic ecology are bounded by the GEIS and no additional mitigation measures beyond those anticipated as conditions of the CSLC easement lease amendment are likely to be warranted.

6. Terrestrial Ecology

The SONGS site is almost entirely paved and developed. However, there are small strips of intact scrub-shrub habitat and ornamental vegetation surrounding the parking lots and between developed areas of the plant. The SONGS site also has undeveloped coastal bluffs that are explicitly protected from development under the CCC Guarantee Agreement. The onsite coastal bluff in the northwest area of SONGS is sparsely vegetated, California desert-thorn scrub habitat. The larger onsite coastal bluff in the southeast area of SONGS is approximately 5 acres and is dominated by California sagebrush scrub vegetation. This bluff is contiguous with the San Onofre bluffs of the San Onofre State Beach, which supports two native vegetation associations (Diegan coastal sage scrub and southern foredune) and small areas of disturbed coastal sage scrub habitat. The coastal bluff areas provide opportunity to support wildlife; however, the light, noise, and frequent human presence due to the proximity of SONGS and the state beach result in a more disturbed habitat than will otherwise be optimal for many species. Avian species are highly mobile and not subject to barriers such as roads and developed areas and may utilize scrub habitat or open surfaces for nesting and temporary perching.

The decommissioning activities will include noise and dust from dismantlement of facilities and heavy equipment traffic, surface runoff, emissions from construction equipment, and the potential for bird interactions with crane booms or other construction equipment. These activities will be conducted in compliance with air quality and noise regulations, and SCE will use avoidance and minimization measures to address potential impacts. Compliance with applicable regulations, air permits, noise restrictions along with the temporary nature of the various decommissioning tasks (e.g., use of cranes) will minimize the impacts to terrestrial species as well as the human community. Decommissioning plans do not currently include the use of explosives, which could disturb terrestrial resources. Should those plans change the environmental impacts will be reevaluated.

SONGS is located within the coastal zone and prior to active dismantlement, SCE will file a coastal development permit application with the CCC. As part of this permitting process, decommissioning activities within the coastal sage habitat areas, coastal bluff, and beach areas will be reviewed by the CCC and United States Fish and Wildlife Service (USFWS) for potential environmental impacts including

the federally listed coastal California gnatcatcher and other protected species and species of concern. Any necessary mitigation measures will be included as conditions of the CCC permit. The removal of various current SONGS features along the perimeter of the developed plant adjacent to and within the natural area could potentially require ground disturbance in unpaved areas. Appropriate avoidance and minimization measures will be used to minimize the impact of any ground disturbance.

With the implementation of appropriate avoidance and minimization measures and compliance with permit conditions as discussed above, decommissioning of SONGS Units 2 and 3 is not anticipated to adversely impact any terrestrial resources and the impacts will be bounded by the GEIS which determined them to be SMALL.

7. Threatened and Endangered Species

Seventeen federally or state protected species utilize habitat within the vicinity (a 6-mile radius) of the SONGS site. These species are listed in Table IV-1, along with their protection status and critical habitat designation. Other species of concern are also addressed in the detailed Environmental Impact Evaluation including both the critically imperiled and imperiled species listed in the California Natural Diversity Data Base and located within one mile of the site but are not otherwise addressed here.

The list includes four federally listed marine turtles. However, none is considered a full-time resident in the vicinity of SONGS and they only migrate through the vicinity. Another federally listed marine reptile, the Hawksbill turtle, sporadically nests in the southern part of the Baja peninsula and foraging sub-adults and juveniles have been sighted along the California coast. Given the SMALL impacts on water use and water quality during decommissioning and the ability of these species to migrate away from the site, these species should not be adversely impacted by decommissioning.

The decommissioning activities will indirectly impact protected species through dust generation from structure demolition, noise from dismantlement of facilities and heavy equipment traffic, surface runoff, emissions from construction equipment, and potential bird interactions with crane booms or other construction equipment. The decommissioning activities will be conducted in compliance with air quality and noise regulations and SCE will use appropriate avoidance and minimization measures. Compliance with applicable regulations, air permits, and noise restrictions related to daylight working along with the temporary nature of the various decommissioning tasks will minimize any such impacts. Decommissioning plans do not currently include the use of explosives, which could disturb protected species. These measures will minimize impacts to protected terrestrial species that inhabit or visit the SONGS site.

Although rare on the site, there has historically been one protected plant species in the vicinity of SONGS, the thread-leaved brodiaea. Decommissioning activities will generally be confined to previously disturbed areas (e.g., paved, high traffic areas). Otherwise, the SCE environmental staff will conduct an environmental assessment per established procedures. The procedure requires an assessment prior to any land disturbance, soil addition, digging, grading, or trenching outside the paved and concreted areas; maintenance activities near surface water, and wetlands and trimming or removal of native plants

other than landscape maintenance. Therefore, adverse impacts on protected plant species are not anticipated.

Decommissioning of SONGS Units 2 and 3 is not anticipated to adversely impact any federally or state-listed species. As discussed above, decommissioning activities will generally be limited to previously disturbed areas on-site, near-shore and off-shore. SCE will employ mitigation measures as required by the regulatory agencies to minimize impacts to the environment and protect listed species. In addition, SCE will implement BMPs and conduct assessments as called for in its environmental protection procedure(s), as well as comply with permit and regulatory requirements to minimize indirect impacts from noise, air emission, dust, and runoff. Therefore, impacts to threatened or endangered species from decommissioning are expected to be SMALL.

Table IV-1

Threatened and Endangered Species Identified within the Vicinity of SONGS

Scientific Name	Common Name	State Status[a]	Federal Status[b]	Critical Habitat within Vicinity
AMPHIBIAN SPECIES				
Anaxyrus californicus	Arroyo toad	—	FE	yes[c]
AVIAN SPECIES				
Charadrius alexandrinus nivosus	Western snowy plover	—	FT	yes[c]
Empidonax traillii extimus	Southwestern willow flycatcher	SE	FE	No
Haliaeetus leucocephalus	Bald eagle	SE	delisted	No
Polioptilacalifornica californica	Coastal California gnatcatcher	—	FT	yes[c]
Vireo bellii pusillus	Least Bell's vireo	SE	FE	yes[c]
FISH SPECIES				
Orcorhynchus mykiss	Steelhead trout	—	FE	yes[c]
INVERTEBRATE SPECIES				
Branchinecta sandiegoensis	San Diego fairy shrimp	—	FE	yes[c]
Streptocephalus woottoni	Riverside fairy shrimp	—	FE	No
MAMMALIAN SPECIES				

Scientific Name	Common Name	State Status[(a)]	Federal Status[(b)]	Critical Habitat within Vicinity
Dipodomys stephensi	Stephen's kangaroo rat	ST	FE	No
Perognathus longimembris pacificus	Pacific pocket mouse	—	FE	No
PLANT SPECIES				
Brodiaea filifolia	Thread-leafed brodiaea	SE	FT	yes[(c)]
REPTILIAN SPECIES				
Caretta caretta	Loggerhead sea turtle	—	FE	No
Chelonia mydas	Green sea turtle	—	FT	No
Dermochelys coriacea	Leatherback sea turtle	—	FE	No
Lepidochelys olivacea	Olive Ridley's turtle	—	FT	No

a. SE = state endangered; ST = state threatened;
b. FE = federally endangered; FT = federally threatened
c. The USFWS has critical habitat delineated within the SONGS site vicinity. However, the designation explicitly excludes Camp Pendleton and thus the SONGS site. Further, the term vicinity includes any area within a 6 mile radius of the site and is not limited to the site itself.

8. Radiological

Decommissioning activities have the potential to contribute to radiological impacts. SONGS Units 2 and 3 may continue to have limited gaseous and liquid radiological effluents until most of the decommissioning activities are complete and the irradiated fuel is transferred to dry storage. SCE is evaluating options to significantly reduce, if not eliminate, routine liquid effluents through the use of self-contained clean-up systems for ongoing systems and activities.

Occupational Dose
The GEIS estimates for the reference pressurized water reactor (PWR) dose is 1,215 person-rem for DECON. In the most recent supplement to the GEIS, the NRC reviewed data available from decommissioning experience subsequent to their initial review (in 1988). Because the range of cumulative occupational doses reported by reactors undergoing decommissioning was similar to the range of estimates for reference plants presented in the 1988 revision of the GEIS, the NRC did not update its estimates for occupational dose.

SCE expects the SONGS dose to be bounded by the referenced PWR dose since: a number of major components which often contribute to area dose rates are relatively new (steam generators and reactor vessel head); and, as a result of SONGS operational dose reduction efforts (i.e., zinc injection). A more detailed estimate will be developed to support evaluation of decontamination scope.

The regulatory standard for worker exposure is a dose limit per worker rather than a cumulative dose. Detailed occupational dose estimates will be performed as part of the work planning process. Such planning will address means to reduce occupational dose where appropriate. SCE remains committed to keeping dose to plant personnel 'As Low as Reasonably Achievable' (ALARA). The activities that have potential radiological impacts will be conducted in a manner to keep doses ALARA and well within regulatory limits.

Public Dose
The NRC generically concluded that reactors undergoing decommissioning could reasonably be expected to have emissions and public doses comparable to or substantially less than the levels experienced during normal operation of those facilities. The Radiological Environmental Monitoring Program (REMP) results demonstrate that the radiological environmental impact of the operation of SONGS Units 2 and 3, and the resulting dose to a member of the general public, is negligible.

SCE will continue to monitor effluents, comply with all applicable regulatory limits, and continue its REMP to assess the impacts to the environment from these effluents.

In summary, SCE estimates that SONGS Units 2 and 3 decommissioning activities will result in occupational and public doses within NRC estimates. Therefore, SONGS' radiological impacts during decommissioning are bounded by the GEIS which determined the radiological impacts to be SMALL.

9. Radiological Accidents

Many activities that occur during decommissioning are similar to activities that commonly take place during maintenance outages at operating plants such as decontamination and equipment removal. Accidents that could occur during these activities may result in injury and local contamination. However, they are not likely to result in contamination off-site.

The limiting design basis accidents (DBAs) applicable to a decommissioning plant are those involving the spent fuel pool. All DBAs and severe accidents involving the reactor are precluded as a result of transfer of spent fuel from the reactor vessels to the pools and ultimately the ISFSI. The environmental impacts of DBAs, including those associated with the spent fuel pool, were evaluated during the initial licensing process and documented in the FES. Furthermore, the impacts of these events are less than previously evaluated due to the time since the fuel was most recently irradiated.

The NRC's GEIS analysis relies in part on the waste confidence rule regarding spent nuclear fuel related severe accidents. The waste confidence GEIS (Reference 9) continues to consider severe accidents involving the spent fuel pool to be a SMALL risk.

Thus, SONGS' radiological accident impacts during decommissioning are bounded by NRC's Decommissioning GEIS which determined such risks to be SMALL.

10. Occupational Issues

SONGS currently has an industrial safety program and safety personnel to promote safe work practices and respond to occupational injuries and illnesses. Equivalent safety programs will continue to be in effect during decommissioning activities.

SONGS has an average occupational injury rate well below that of the heavy construction industry sector and consistent with the power generation and nuclear power industry. Decommissioning activities will be conducted in a manner reflecting personnel safety as a critical element. Therefore, SONGS occupational safety impacts are considered to be bounded by the GEIS which generically determined occupational safety impacts to be SMALL.

11. Cost

Decommissioning costs for SONGS are discussed in the DCE being submitted concurrently.

12. Socioeconomics

The primary socioeconomic impacts of decommissioning are related to staffing changes and decreasing tax revenues. Impacts related to the decision to permanently cease operations are outside the scope of this evaluation. SCE determined the staff reduction impacts from the decision to be minimal. The staff reductions represent 0.04 percent and 0.03 percent of San Diego County's and Orange County's workforces, respectively. Any impacts will be deferred somewhat due to the employment of temporary staff necessary to accomplish the various decommissioning activities.

Similarly, SONGS is located in San Diego County and its property assessment is a relatively small portion of San Diego County's total tax collections. Historically, SONGS' contribution to the county property tax collections has been consistently less than 1 percent. SONGS' tax obligations will be reduced due to decommissioning, but SCE and SONGS will continue to contribute to county tax revenues.

It is anticipated that there will be limited or no changes or impacts to the local community and socioeconomic conditions and less impact than would be expected generically where other nuclear facilities have a higher relative impact on the job market or tax base. Thus, SONGS' impacts are bounded by those considered in the GEIS in which the NRC generically determined socioeconomic impacts to be SMALL.

13. Environmental Justice

Decommissioning activities that may potentially affect identified minority and low-income populations include those related to staffing changes and offsite transportation. However, the assessment of environmental justice also considered other specific issues (e.g., water use, air quality). SCE has determined that no significant offsite impacts will be created by SONGS 2 & 3 decommissioning activities. As generic NRC guidance recognizes, if no significant offsite impacts occur in connection with the proposed action, then no member of the public will be substantially affected. Therefore, there can

be no disproportionately high and adverse impacts on members of the public, including minority and low-income populations. In addition, staffing is not anticipated to be an impact due to the large population and robust job market in the area (see Section 12 above).

The environmental justice evaluations utilize a 50-mile radius around the plant as the potentially impacted area. To complete this evaluation, the 2006–2010 low-income data and 2010 minority population data for California were obtained from the United States Census Bureau (USCB) and processed using ESRI ArcGIS 10.1 software. All census data were downloaded in USCB block group level geography so that the environmental justice evaluations were consistent between the minority and low-income analyses. The evaluations and results are detailed in the EIE which concluded there were no disproportionate impacts.

In its GEIS, the NRC concluded that adverse environmental justice impacts and associated significance of the impacts must be determined on a site-specific basis. Unlike many nuclear sites, SONGS is located in and near relatively large communities with significant other commercial and industrial activities. Thus, the impact of SONGS shutdown is less severe than may otherwise be the case. Further, SCE has determined that no significant offsite environmental impacts will be created by SONGS Units 2 and 3 decommissioning activities. Since no significant offsite impacts occur in connection with the proposed action, no member of the public will be substantially affected. Therefore, it is unlikely for there to be a disproportionately high and adverse impact or effects on specific groups or members of the public, including minority and low-income populations, resulting from the decommissioning of SONGS Units 2 and 3.

14. Cultural Historic and Archeological Resources

No prehistoric or historic archaeological sites or historic sites eligible for listing or listed on the National Register of Historic Places, California Register of Historical Resources, or San Diego County Local Register of Historical Resources are located within the SONGS site lease easement and no traditional cultural properties are known to be present. Two prehistoric archaeological sites and three historic archaeological sites were identified within 0.5 miles of SONGS Units 2 and 3.

All of these areas are outside the operational/decommissioning site. In its GEIS, the NRC concluded that for plants where the disturbance of lands beyond the operational areas is not anticipated, the impacts on cultural, historic, and archeological resources will be SMALL. Since decommissioning activities are confined to the SONGS site, no adverse impacts are anticipated. SONGS' impacts on cultural, historical, and archeological resources during decommissioning fall well within the bounds established by the NRC in the GEIS.

15. Aesthetic Issues

In its GEIS, the NRC stated that removal of structures is generally considered to be a beneficial aesthetic impact and drew the generic conclusion that for all plants, the potential impacts from decommissioning on aesthetics are SMALL and that any mitigation measures are not likely to be beneficial enough to be

warranted. Similarly, the aesthetic impact of final result of decommissioning SONGS Units 2 and 3 will be less than that of the current aesthetic impact of the plant. During dismantlement, any adverse visual intrusion will be temporary and will ultimately serve to reduce the aesthetic impact of the site. Therefore, the impacts of SONGS on aesthetic resources during decommissioning are bounded by the GEIS.

16. Noise

Offsite noise sources that affect the ambient noise environment in the vicinity of SONGS include Interstate-5, the San Diego Northern Railroad, and military operations. During the decommissioning process, the sounds that might be heard at offsite locations include noise from construction vehicles and tools. The timing of noise impacts and the duration or intensity will vary. The nearest sensitive receptors to SONGS are recreational users of San Onofre State Beach where the ambient noise environment can exceed 70 dBA. The more intense decommissioning activities will occur 400 ft or more from the beach access public walkway in front of the SONGS sea wall.

Due to the relatively high ambient noise levels surrounding SONGS, decommissioning activities are not expected to produce noise levels that could impact the activities of humans or threatened and endangered species. In addition, SCE will comply with the local noise regulations for construction sites, which restrict the average sound level at the property boundary to 75 dBA between 7 a.m. and 7 p.m., and any additional agency permit requirements including any lower allowed limits during evenings and overnight. Therefore, noise impacts during decommissioning of SONGS Units 2 and 3 are bounded by the previously issued GEIS, which generically determined the noise impacts associated with decommissioning to be SMALL.

17. Transportation

Transportation impacts are dependent on the number of shipments to and from the facility, the type of shipments, the distance that material is shipped, and the number of workers commuting to and from the site.

Transportation infrastructure within the vicinity of SONGS includes one major north- and south-bound freeway, I-5, an assortment of local and county roads, passenger and cargo rail service (part of the Los Angeles–San Diego corridor), and an existing rail spur serving the SONGS site. The 2011 average annual daily traffic (AADT) count for this portion of I-5 was 132,000 vehicles.

SCE compared the assumptions and analysis inputs used for NRC's analysis with waste volumes estimated for SONGS Units 2 and 3 decommissioning, transport mode, and disposal facility options. Due to the availability of the rail line, a substantial portion of the shipments will likely use that mode of transportation. The NRC indicates use of rail reduces radiological impacts by more than a factor of 10 over truck shipments. Furthermore, disposal facilities available for SONGS Units 2 and 3 radiological wastes are less than half the distance assumed by NRC in its analysis. Therefore the generic impacts bound those associated with SONGS Units 2 and 3.

Furthermore, SCE will comply with all applicable NRC and U.S. Department of Transportation (DOT) regulations, including Federal Railroad Administration regulations and requirements, and will use approved packaging and shipping containers for waste shipment. SCE will also comply with State of California regulations enforced by Caltrans and the California Highway Patrol. The NRC has generically concluded that the radiological impacts of transporting radiological waste from decommissioning will be SMALL and those for SONGS Units 2 and 3 are bounded by the GEIS.

SCE estimated a peak of approximately 560 workers during decommissioning and the vehicular traffic due to commuting will likely exceed the 200 per peak hour threshold, prompting review for potential to impact traffic congestion as required under the local congestion management plan. SCE estimated peak truck traffic due to waste shipments to be approximately 150 per day. The decommissioning traffic associated with SONGS is considered negligible compared to existing traffic volumes and will not be expected to significantly alter congestion on roadways. In addition, this amount of traffic is not expected to significantly deteriorate roadways; therefore the GEIS is bounding and the non-radiological transportation impacts of decommissioning are SMALL.

Offshore activities to remove vertical risers on the intake and discharge conduits will increase marine vessel traffic in the area. It is expected that these activities will not cause either a navigational safety hazard or a substantial delay in the normal movements of commercial or recreational vessels. The environmental impacts review for the Unit 1 conduit disposition indicated that impacts to recreational and commercial transportation will be insignificant.

18. Irreversible and Irretrievable Commitment of Resources

SONGS Units 2 and 3 decommissioning will involve dismantlement and removal of structures and restoration of the property to a state for unrestricted release per NRC regulations in accordance with the criteria for license termination in 10 CFR 20, Subpart E. Furthermore, the property will be returned to the U.S. Navy under negotiated terms of the easement. The activities necessary to decommission SONGS Units 2 and 3 involve a minor irretrievable commitment of consumable materials (including materials for decontamination, solvents, industrial gases, tools, fuel, etc.). The irreversible commitment of such resources is not unique and is bounded by those considered by the NRC in the GEIS which concluded consumption to be minor.

Waste from decommissioning of SONGS Units 2 and 3 will consume space at waste facilities. California has multiple facilities permitted for the storage, treatment, and disposal of hazardous and universal waste. The nonradioactive waste is assumed to be shipped to an out-of-state landfill due to the moratorium on disposal of decommissioned materials at California nonhazardous landfills. The decommissioning of SONGS Units 2 and 3 will result in minor irretrievable or irreversible commitment of resources bounded by the GEIS in which the NRC determined will be SMALL impacts.

B. Environmental Impacts of License Termination – NUREG-1496

The License Termination Plan (LTP) has not yet been developed. As noted earlier, it is required to be submitted at least two years prior to the proposed termination date. In general, the LTP outlines the basis for an administrative/legal activity. No physical work beyond that already addressed is anticipated. Thus, there are no environmental impacts beyond those already addressed that need to be addressed at this point in the process.

C. Discussion of Decommissioning in the FES

Applicable portions of the FES were addressed as noted in each of the topics previously summarized.

D. Additional Considerations

SCE has not identified any unique considerations that need to be further addressed. The previous topic summaries address a sufficiently wide range of issues.

E. Conclusion

SCE has performed an environmental review to evaluate environmental impacts associated with decommissioning activities, confirming that the anticipated or potential impacts are within the bounds of the generic impacts that NRC described in the GEIS. Further, while there are no applicable bounding impacts for threatened and endangered species and environmental justice discussed in the GEIS, the SONGS Units 2 and 3 decommissioning activities are not anticipated to result in significant impacts to threatened and endangered species or disproportionate impacts on minority or low-income populations. This is principally due to the following:

- Planned activities fall within the activities that the NRC evaluated. There are no unique aspects of the plant or decommissioning techniques that will invalidate previously drawn conclusions.
- Methods to be employed to dismantle and decontaminate the site are standard construction-based techniques fully considered in the GEIS.
- SCE will continue to comply with NRC dose limits and conduct activities in accordance with ALARA principles.
- SCE will continue to comply with the SONGS Offsite Dose Calculation Manual, Radiological Effluent Monitoring Program, and the Ground Water Protection Initiative Program during decommissioning. Each will likely be modified somewhat to reflect changes in site configuration, etc.
- SCE will comply with all applicable NRC and DOT regulations, including Federal Railroad Administration regulations and requirements, and use approved packaging and shipping containers for the shipping of radiological waste. SCE will also comply with State of California regulations enforced by Caltrans and the California Highway Patrol.

- SCE will continue to comply with federal, state, and local requirements for non-radiological interfaces with the environment including limitations on water withdrawal and discharges, air emissions including criteria pollutants and fugitive dust, noise levels, protection of avian, terrestrial and aquatic species, cultural resources, disposal of non-radiological waste, and worker health protection.
- SCE will seek and comply with an amendment to its CSLC easement lease to largely abandon the intake and discharge conduits in place.
- SCE will seek and comply with a coastal development permit from the CCC for decommissioning.

V. REFERENCES

A. GENERAL DEVELOPMENTAL REFERENCES

1. NRC Regulatory Guide 1.185, Revision 1, June 2013, Standard Format and Content Guide for Post-Shutdown Decommissioning Activities Report
2. Energy*Solutions* Document No. 164001, "2014 Decommissioning Cost Analysis of the San Onofre Nuclear Generating Station Units 2 and 3"
3. Enercon Technical Data Record No. SONGS002, "SONGS Units 2 and 3 Environmental Impact Evaluation"

B. SPECIFIC REFERENCES IN TEXT

1. Letter from Thomas J. Palmisano (SCE) to the U. S. Nuclear Regulatory Commission dated February 13, 2014; Subject: Access to Nuclear Decommissioning Trust Funds, San Onofre Nuclear Station, Units 2 and 3.
2. Letter from Richard C. Brabec (SCE) to the U. S. Nuclear Regulatory Commission dated March 31, 2014; Subject: Decommissioning Funding Status Report, San Onofre Nuclear Generating Station Units 2 and 3
3. Letter from P. T. Dietrich (SCE) to the U. S. Nuclear Regulatory Commission dated June 12, 2013; Subject: Certification of Permanent Cessation of Power Operations San Onofre Nuclear Generating Station, Units 2 and 3
4. Letter from P. T. Dietrich (SCE) to the U. S. Nuclear Regulatory Commission dated June 28, 2013; Subject: Permanent Removal of Fuel from the Reactor Vessel, San Onofre Nuclear Generating Station Unit 3
5. Letter from P. T. Dietrich (SCE) to the U. S. Nuclear Regulatory Commission dated July 22, 2013; Subject: Permanent Removal of Fuel from the Reactor Vessel, San Onofre Nuclear Generating Station Unit 2
6. U. S. Nuclear Regulatory Commission; NUREG-0586, "Final Generic Environmental Impact Statement (GEIS) on Decommissioning Nuclear Facilities" (November 2002)
7. AIF/NESP-036, "A Guideline for Producing Commercial Nuclear Power Plant Decommissioning Cost Estimates"
8. U.S. Nuclear Regulatory Commission, NUREG-0490, "Final Environmental Statement related to the operation of San Onofre Nuclear Generating Station, Units 2 and 3" (April 1981)
9. U. S. Nuclear Regulatory Commission, NUREG-2157, "Waste Confidence Generic Environmental Impact Statement, Report for Comment" (August 2014)
10. U. S. Nuclear Regulatory Commission, NUREG-1496, Volume 1, "Generic Environmental Impact Statement in Support of Rulemaking on Radiological Criteria for License Termination of NRC-Licensed Nuclear Facilities" (July 1997)
11. NEI 07-07, "Industry Groundwater Protection Initiative, Final Guidance Document," in August 2007

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Thomas J. Palmisano
Vice President & Chief Nuclear Officer

10 CFR 50.82(a)(4)(i)

September 23, 2014

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington D.C. 20555-0001

Subject: **Docket Nos. 50-361 and 50-362, San Onofre Nuclear Generating Station, Units 2 and 3 Irradiated Fuel Management Plan**

Reference Letter from P.T. Dietrich (SCE) to the U.S. Nuclear Regulatory Commission, dated June 12, 2013; Subject: Certification of Permanent Cessation of Power Operations, San Onofre Nuclear Generating Station, Units 2 and 3

Dear Sir or Madam:

On June 12, 2013, SCE submitted the referenced letter to the U.S. Nuclear Regulatory Commission (NRC) certifying the permanent cessation of operations at San Onofre Nuclear Generating Station (SONGS), Units 2 and 3, in accordance with 10 CFR 50.54(bb) and 10 CFR 50.82(a)(4)(i), Southern California Edison (SCE) is required to submit an Irradiated Fuel Management Plan (IFMP), Site Specific Decommissioning Cost Estimate (DCE) and Post-Shutdown Decommissioning Activities Report (PSDAR) within two years of permanent cessation of operations.

The SONGS, Units 2 and 3 IFMP is attached. The DCE and PSDAR are being concurrently submitted under separate cover letters. The IFMP represents SCE's current plans and is subject to change as the project progresses. In particular, the Independent Spent Fuel Storage Installation location, and storage equipment and vendor(s) have not been selected. The decision making and procurement activities are underway but have not been finalized.

Changes to significant details will be included in subsequent revisions to the IFMP as required by 10 CFR 50.54(bb). Financial assurance information will be provided on an annual basis as required by 10 CFR 50.75(f)(1).

This letter does not contain any new commitments.

If there are any questions or if additional information is needed, please contact me or Ms. Andrea Sterdis at (949) 368-9985.

Sincerely,

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A001
NRR

Enclosure: San Onofre Nuclear Generating Station Units 2 and 3 Irradiated Fuel Management Plan

cc: M. L. Dapas, Regional Administrator, NRC Region IV
T. J. Wengert, NRC Project Manager, SONGS, Units 2 and 3
T. J, Warnick, NRC Project Manager, San Onofre Units 2 and 3 Decommissioning
R. E. Lantz, NRC Region IV, San Onofre Units 2 and 3
S. Y. Hsu, California Department of Health Services, Radiologic Health Branch

SONGS Units 2 and 3 Irradiated Fuel Management Plan

I. Background and Introduction

On June 12, 2013, Southern California Edison (SCE) submitted a letter to the U.S. Nuclear Regulatory Commission (NRC) (Reference 1) certifying the permanent cessation of operations at San Onofre Nuclear Generating Station (SONGS) Units 2 and 3 effective June 7, 2013, in accordance with 10 CFR 50.82(a)(1)(i). All fuel was removed from the SONGS Units 2 and 3 reactor vessels and placed in their respective spent fuel pools as certified in accordance with 10 CFR 50.82(a)(1)(ii) (References 2 and 3).

Pursuant to 10 CFR 50.54(bb), licensees are required to submit a plan for the management of irradiated fuel until title and possession of the fuel is transferred to the Secretary of Energy for its ultimate disposal in a repository. The Irradiated Fuel Management Plan (IFMP) is required to be submitted to the Commission either five years before expiration of the Operating License or within two years following permanent cessation of operations, whichever occurs first. Therefore, the SONGS Units 2 and 3 plans are required to be submitted prior to June 7, 2015, two years following the cessation of operations. This submittal constitutes SCE's IFMP for SONGS Units 2 and 3, submitted on behalf of itself and the other SONGS Participants responsible for the funding of the SONGS decommissioning. The other SONGS Participants are the City of Anaheim, the City of Riverside, and San Diego Gas & Electric Company (SDG&E).

Energy*Solutions*, LLC has prepared a site-specific decommissioning cost estimate (DCE) for SONGS Units 2 and 3 (Reference 15). The DCE identifies the details, schedules, and costs of spent fuel management activities associated with the IFMP, along with license termination and site restoration activities and costs. This DCE is being submitted concurrent with the Post-Shutdown Decommissioning Activities Report (PSDAR, Reference 4) and this plan. The assumptions regarding the United States Department of Energy (US DOE) acceptance of irradiated fuel is consistent with the Energy*Solutions* DCE and is based on testimony filed with the California Public Utility Commission (Reference 13). The SONGS Units 2 and 3 DCE and this IFMP are based on commencement of industry-wide acceptance of spent fuel by US DOE in 2024.

II. Irradiated Fuel Management Strategy

The safe initial interim storage of SONGS Units 2 and 3 irradiated fuel will be "wet storage" in each unit's respective spent fuel pool. The spent fuel pools will be isolated from their normal support systems and those systems replaced by stand-alone cooling and filtration units (also termed a "spent fuel pool island"). Doing so facilitates earlier system abandonment and parallel decommissioning activities.

Subsequently, all irradiated fuel in the SONGS Units 2 and 3 spent fuel pools will be safely transferred to "dry storage" at the common Independent Spent Fuel Storage Installation (ISFSI) located on the SONGS site. Dry storage is also considered interim storage pending transfer to the US DOE.

A total of 1,726 irradiated fuel assemblies have been generated in SONGS Unit 2 and 1,734 irradiated fuel assemblies have been generated in SONGS Unit 3, for a total of 3,460 irradiated fuel assemblies. At present, 792 SONGS Units 2 and 3 irradiated fuel assemblies have already been transferred to the common ISFSI. The remaining 2,668 irradiated fuel assembles will be loaded into Dry Shielded Canisters (DSCs) and transferred to the ISFSI.

The current ISFSI is located inside the Owner Controlled Area. It was constructed to accommodate SONGS Unit 1 irradiated fuel and provides additional capacity for a limited amount of SONGS Units 2 and 3 irradiated fuel.

The ISFSI currently contains 18 DSCs storing Unit 1 fuel and Greater than Class C (GTCC) waste. The ISFSI also contains 33 DSCs which store Units 2 and 3 fuel. All of the fuel on the ISFSI is stored in Transnuclear NUHOMS Model Number-24PT1 or PT4 DSCs.

The major IFMP activity phases, including start and end dates and associated costs for each period are identified in Table 1. The identified Spent Nuclear Fuel (SNF) Periods are developed in and align with the site-specific DCE (Reference 15).

The current plans are to obtain necessary permits for the ISFSI to be expanded to accommodate the remaining inventory of the SONGS Units 2 and 3 spent fuel pools. SONGS plans to commence the movement of irradiated fuel from the Unit 2 and Unit 3 pools to the ISFSI in 2017. SONGS expects to complete the transfer in 2019. Additional DSCs will be procured from one or more of the available dry storage system suppliers beginning in 2014. An additional 47 DSCs will be required for the SONGS Unit 2 irradiated fuel and an additional 44 DSCs will be required for the SONGS Unit 3

irradiated fuel (depending on the capacity of the selected system and the number of DSCs needed to store GTCC waste and other materials). The spent fuel pool inventory is forecast to be transferred to the ISFSI no later than the end of 2019.

The US DOE Standard Contracts for acceptance and disposal of spent nuclear fuel and high level waste contain the basis for the initial ranking of industry-wide spent fuel acceptance obligations based upon the date of permanent removal of the spent nuclear fuel from service ("oldest fuel first" allocation). Those Standard Contracts also contain provisions allowing for "exchanges" of acceptance obligations, and priority for retired units. Given the US DOE's lack of performance, a common assumption for purposes of this fuel management plan is to base acceptance projections upon application of an "oldest fuel first" allocation scheme to a projected start date for repository operations. This plan is based upon a 2024 start date (Reference 13) for US DOE acceptance of spent fuel from the industry and the SONGS Units 2 and 3 positions in the queue. As indicated in Table 3, SCE is therefore assuming all fuel will be removed from the SONGS site as of 2049. Based on this assumption, the ISFSI will be subsequently decommissioned by the 2051 final license termination date.

III. Financial Assurance

The regulations (10 CFR 50.54(bb)) also require that funding adequacy be demonstrated to support the irradiated fuel management plan.

The cost of twelve (12) additional DSCs to be stored on the current ISFSI was funded from sources other than the Nuclear Decommissioning Trusts (NDT) (Reference 5), as are the costs associated with ongoing storage of Unit 1 spent fuel at the GE-Hitachi Nuclear America LLC's Morris Operation ISFSI located in Morris, Illinois. Table 1 includes the costs of procurement and construction of the expanded ISFSI capacity and all loading costs. Operation of the spent fuel pools is modeled as being discontinued in 2019 after all of the fuel has been transferred to dry storage. ISFSI operations continue until the US DOE is able to complete the transfer of the SONGS fuel to a repository or interim storage facility, which is currently assumed to occur by 2049.

SONGS management is committed to providing consistent and up-to-date information to all of its stakeholders and regulators. Aspects of the SONGS Nuclear Decommissioning Trust Fund are regulated by both the California Public Utilities Commission (CPUC) and the NRC. Previous Decommissioning Cost Estimates (DCEs) were updated and submitted to the CPUC as part of the Nuclear Decommissioning Cost Triennial Proceedings (Reference 5). Financial assurance reports including the balances and expenditures for SONGS Unit 1 were supplied to the NRC (as required by 10 CFR

50.82(a)(8)(v)) annually (most recently in Reference 6) and balances for SONGS Units 2 and 3 were submitted on a biennial basis (as required by 10 CFR 50.75(f)(1)) (most recently in Reference 7). Reports regarding ISFSI costs and decommissioning funding assurance for these costs were summarized triennially as required by 10 CFR 72.30(c) (most recently in Reference 8). Going forward, balances and expenditures will be supplied annually to the NRC for all three units and the ISFSI.

An updated site-specific DCE will be concurrently submitted to the NRC. As summarized in Table 1, this plan is based on decommissioning and the termination of the license by 2051, approximately 38 years following the permanent cessation of operations. The summary in Table 1 includes the funds for dry storage through 2049 and final release of the ISFSI in 2051.

Tables 4A and 4B summarize the estimated annual spending for all decommissioning activities (License Termination, Spent Fuel Management, and Site Restoration), and combined NDT current balances in 2014 dollars. Table 2 reflects key tasks addressed by the NRC staff in a recent safety evaluation.

The total of all Nuclear Decommissioning Trust funds balances for SONGS Units 2 and 3 was $3,926 million as of December 31, 2013 (Reference 9). Evaluation of the projected cash flows assuming earnings on existing balances as permitted by NRC regulations demonstrates the adequacy of the existing funds to cover all aspects of decommissioning, including the costs of irradiated fuel management. This demonstrates that the balance in the decommissioning trust is adequate to fund all aspects of decommissioning as well as the costs of irradiated fuel management. As decommissioning proceeds the DCE will be updated as appropriate and annual updates of spending and trust fund balances will be docketed as required.

IV. <u>Regulatory Activities</u>

The IFMP assumes that the SONGS Participants will make withdrawals from their nuclear decommissioning trusts for spent fuel management purposes. The SONGS Participants have collected funds from ratepayers and accumulated funds in the nuclear decommissioning trusts for the purpose of funding three primary categories of costs: (1) License Termination; (2) Spent Fuel Management; and (3) Site Restoration. On November 18, 2013, SCE filed a Tier 3 Advice Letter (Reference 10) with the CPUC to obtain authorization for the use of funds in the near term and to establish processes for further CPUC oversight of withdrawals from the nuclear decommissioning trusts. On February 21, 2014, SDG&E filed a similar letter (Reference 14) with the CPUC. In addition to authorizing and overseeing the withdrawals, the CPUC is expected to

designate the specific amounts from the existing fund balances that are available for License Termination and therefore subject to 10 CFR 50.82(a)(8)(i)(A) and 10 CFR 50.75(h)(2). The fund balances would then be allocated to separate subaccounts within each trust fund and, as such, available for spent fuel management and site restoration, consistent with the requirements of 10 CFR 50.75, 10 CFR 50.82, and 10 CFR 72.30.

To confirm such access, SCE requested (Reference 11) an exemption from 10 CFR 50.75 and 50.82 to authorize the use of trust funds to pay for spent fuel management and site restoration including other transitional costs. The regulations limit the use of the nuclear trust fund to decommissioning costs. This exemption was granted on September 5, 2014 (Reference 12).

The SONGS Participants responsible for decommissioning will periodically review the amount of cash contributions required for the decommissioning fund to ensure that withdrawals do not inhibit the ability of the licensee to complete NRC License Termination, Spent Fuel Management, and Site Restoration. The SONGS Participants will obtain authorization as necessary through the ratemaking processes to provide for further contributions if required.

In accordance with 10 CFR 50.82(a)(8)(vii), SONGS will annually submit to the NRC by March 31st a report on the status of the funding for managing spent fuel. The report will include, current through the end of the previous calendar year, the amount of funds accumulated to cover the cost of managing the spent fuel, the projected cost of managing spent fuel until title to the fuel and possession of the fuel is transferred to the Secretary of Energy, and if the funds accumulated do not cover the projected cost, a plan to provide additional funding assurance using one of the methods allowed by NRC regulations.

V. References

1. Letter from P. Dietrich, Southern California Edison, to U.S. Nuclear Regulatory Commission, Subject: Dockets 50-361 and 50,362, Certification of Permanent Cessation of Power Operations, San Onofre Nuclear Generating Station Units 2 and 3, dated June 12, 2013

2. Letter from P. Dietrich, Southern California Edison, to U.S. Nuclear Regulatory Commission, Subject: Dockets 50-361 Permanent Removal of Fuel from Reactor Vessel, San Onofre Nuclear Generating Station, Unit 2, dated July 22, 2013

3. Letter from P. Dietrich, Southern California Edison, to U.S. Nuclear Regulatory Commission, Subject: Dockets 50-362 Permanent Removal of Fuel from Reactor Vessel, San Onofre Nuclear Generating Station, Unit 3, dated June 28, 2013

4. SONGS Units 2 and 3 Post-Shutdown Decommissioning Activities Report, San Onofre Nuclear Generating Station

5. Decommissioning Cost Estimate, 2013 Scenario, dated July 11, 2013, ABZ, Incorporated. Used in support of Nuclear Decommissioning Cost Triennial Proceeding, Exhibit SCE-12

6. Letter from Richard C. Brabec, Southern California Edison to U. S. Nuclear Regulatory Commission, Subject: 10 CFR 50.75(f)(1) and 10 CFR 50.82(a)(8)(v-vii) Decommissioning Funding Status Report San Onofre Nuclear Generating Station Unit 1 dated March 31, 2014

7. Letter from Richard C. Brabec, Southern California Edison to U. S. Nuclear Regulatory Commission, Subject: 10 CFR 50.75(f)(1) Decommissioning Funding Status Report, San Onofre Nuclear Generating Station Units 2 and 3 dated March 31, 2014

8. Letter from Douglas R. Bauder, Southern California Edison U. S. Nuclear Regulatory Commission , Subject: 10 CFR 72.30 ISFSI Decommissioning Funding Plan, San Onofre Nuclear Generating Station Units 1, 2 & 3 dated December 14, 2012

9. Letter from Richard C. Brabec, Southern California Edison to U.S. Nuclear Regulatory Commission, Subject: San Onofre Nuclear Generating Station, Units 2 and 3 Access to Nuclear Decommissioning Trust Funds, Supplemental Information, Dated March 12, 2014

10. Letter from Megan Scott-Kakures, Southern California Edison, to Public Utilities Commission of the State of California Energy Division Submitting a Tier 3 Advice Letter Requesting (1) Authorization of Disbursements from the Master Trusts for San Onofre Nuclear Generating Station; (2) Approval of Tier 2 Advice Letter to Process for Future Disbursements; (3) Designation of Trust Amounts Set Aside for License Termination; and (4) Approval of Balancing Account, dated November 18, 2013

11. Letter from Tom J. Palmisano, Southern California Edison, to U. S. Nuclear Regulatory Commission, Subject: San Onofre Nuclear Generating Station Units 2 and 3, Access to Nuclear Decommissioning Trust Funds, dated February 13, 2014

12. Letter from Thomas Wengert, Nuclear Regulatory Commission to Tom J. Palmisano, Southern California Edison, Granting Exemptions from the Requirements of 10 CFR 50, Sections 50.82(a)(8)(i)(A) and 50.75(h)(2) (TAC Nos. MF3544 an MF 3545) dated September 5, 2014

13. Testimony on Nuclear Decommissioning of SONGS 2 & 3 and Palo Verde, exhibit No. SCE-2, dated December 21, 2012

14. Letter from Clay Faber, San Diego Gas & Electric, to Public Utilities Commission of the State of California submitting a Tier 3 Advice Letter Requesting (1) Designation of SONGS 2&3 Costs Incurred During and After June 2013 As Decommissioning Costs Eligible for Payment with Trust Funds; (2) Authorization of Disbursements from the Master Trusts for San Onofre Nuclear Generating Station; (3) Approval of Tier 2 Advice Letter Process for Future Trust Disbursements; (4) Acknowledgement That Funds Have Been Collected From Ratepayers and Have Been Accumulating In The Trusts To Be Used for NRC and Non-NRC Jurisdictional Decommissioning Cost Categories; and (5) Designation of an Allocation of the SDG&E SONGS 2&3 Trusts Among the Major Decommissioning Cost Categories, dated February 21, 2014

15. EnergySolutions Document No. 164001, “2014 Decommissioning Cost Analysis of the San Onofre Nuclear Generating Station Units 2 and 3”

SONGS Units 2 and 3 Irradiated Fuel Management Plan

Table 1

Irradiated Fuel Management Plan – Summary Schedule

Cost and Schedule Summary (2014 Dollars in thousands)							
Spent Fuel 10 CFR 50.54(bb)							
Period No.	**Period Description**	**Start**	**End**	**Years**	**Unit 2 Cost**	**Unit 3 Cost**	**Total Cost**
SNF Pd 1	Spent Fuel Management Transition	6/7/2013	12/31/2013	0.56	$63,891	$66,105	$129,997
SNF Pd 2	Spent Fuel Transfer to Dry Storage	1/1/2014	6/1/2019	5.41	$344,629	$372,193	$716,822
SNF Pd 3	Dry Storage During Decommissioning – Units 1, 2 and 3	6/1/2019	12/5/2031	12.51	$61,425	$61,425	$122,849
SNF Pd 4	Dry Storage Only – Units 1, 2 and 3	12/5/2031	12/31/2035	4.07	$29,383	$29,383	$58,765
SNF Pd 5	Dry Storage Only – Units 2 and 3	12/31/2035	12/31/2049	14.00	$107,326	$107,326	$214,653
SNF D&D Pd 1	ISFSI License Termination	12/31/2049	5/6/2050	0.34	$1,260	$1,260	$2,520
SNF D&D Pd 2	ISFSI Demolition	5/6/2050	9/8/2051	1.34	$15,295	$15,295	$30,590
	Category Total			**38.23**	**$623,209**	**$652,987**	**$1,276,196**

Table 2
Major Fuel Management Tasks

Major Fuel Management Task Direct Costs (Note 1)	**Explanatory or Additional Details**	**Estimate in DCE** (in Thousands)	**Schedule in DCE**
Estimated Costs to isolate spent fuel pools and fuel handling systems	• Estimated cost for Islanding • No additional costs are required for fuel handling systems. Cranes are single-failure proof	$ 22,183 (Note 2)	6/2015
Estimated cost to construct an ISFSI or a combination of wet/dry storage	• ISFSI in operation; so, current costs are for wet/dry combination. • Costs are associated with capacity expansion (pad and associated facility costs, DSCs and HSMs).	$ 396,391 (Note 3)	6/2019
Estimated annual cost for the operation of the selected option	• Operational and maintenance costs are NOT readily separable (fuel storage support vice other demands); but, are included in Table 4 cash flows.	N/A	Ongoing
Estimated cost for preparation, packaging and shipping of fuel to DOE	• Off-site transportation costs are part of contract with US DOE.	$ 6,742 (Note 4)	Thru 12/2049
Estimated cost to decommission the ISFSI	• Funded from both Unit 1 and Units 2&3 Decommissioning Trust Funds.	$ 33,110 (Note 5)	2049-2051
Brief discussion of selected storage method or methods and estimated time frame for these activities	• See Section II for selected methods. • See Table 1 for time frames.	N/A	N/A

Notes:

1. Tasks from NRC Safety Evaluation (SE) on Kewaunee Integrated Fuel Management Plan dated, September 28, 2009, publically available under ADAMS Accession No. ML092321079
2. Cost based on DCE, DECON Pd 2, Items 2.23 through 2.30
3. Cost based on DCE, SNF Pd 2, Items 8.05 through 8.13
4. Cost based on SNF Pd 4 and SNF Pd 5, Item 2.03
5. Cost based on DCE, total of SNF D&D Pd 1 and SNF Pd 2

SONGS Units 2 and 3 Irradiated Fuel Management Plan

Table 3

SONGS Unit 2 & Unit 3
Spent Fuel Shipping Schedule
2024 DOE Acceptance

	On-Site Inventory (Beginning of the Year)				On-Site Transfers (During Year)		Off-Site Transfers (During Year)			
Year	Unit 2 & 3 Fuel Assemblies in Wet Storage	Units 2 & 3 Fuel Assemblies in Dry Storage	Units 2 & 3 Fuel Assemblies in On-Site Storage	Units 2 &3 Canisters in ISFSI	Unit 2 & 3 Fuel Assemblies Transferred to ISFSI	Unit 2 & 3 Canisters Transferred to ISFSI	Unit 2 Assemblies Transferred to DOE	Unit 3 Assemblies Transferred to DOE	Unit 2 & 3 Assemblies Transferred to DOE	Unit 2 & 3 Canisters Transferred to DOE
2014	2668	792	3460	33	0	0	0	0	0	0
2015	2668	792	3460	33	0	0	0	0	0	0
2016	2668	792	3460	33	0	0	0	0	0	0
2017	2668	792	3460	33	768	24	0	0	0	0
2018	1900	1560	3460	57	1536	48	0	0	0	0
2019	364	3096	3460	105	364	13	0	0	0	0
2020	0	3460	3460	118	0	0	0	0	0	0
2021	0	3460	3460	118	0	0	0	0	0	0
2022	0	3460	3460	118	0	0	0	0	0	0
2023	0	3460	3460	118	0	0	0	0	0	0
2024	0	3460	3460	118	0	0	0	0	0	0
2025	0	3460	3460	118	0	0	0	0	0	0
2026	0	3460	3460	118	0	0	0	0	0	0
2027	0	3460	3460	118	0	0	0	0	0	0
2028	0	3460	3460	118	0	0	0	0	0	0
2029	0	3460	3460	118	0	0	0	0	0	0
2030	0	3460	3460	118	0	0	48	48	96	4
2031	0	3364	3364	114	0	0	192	96	288	12
2032	0	3076	3076	102	0	0	120	120	240	10
2033	0	2836	2836	92	0	0	0	96	96	4
2034	0	2740	2740	88	0	0	112	120	232	8
2035	0	2508	2508	80	0	0	96	96	192	6
2036	0	2316	2316	74	0	0	128	96	224	7
2037	0	2092	2092	67	0	0	0	0	0	0
2038	0	2092	2092	67	0	0	96	128	224	7
2039	0	1868	1868	60	0	0	96	96	192	6
2040	0	1676	1676	54	0	0	96	96	192	6
2041	0	1484	1484	48	0	0	0	0	0	0
2042	0	1484	1484	48	0	0	96	96	192	6
2043	0	1292	1292	42	0	0	96	96	192	6
2044	0	1100	1100	36	0	0	96	96	192	6
2045	0	908	908	30	0	0	128	96	224	7
2046	0	684	684	23	0	0	96	128	224	7
2047	0	460	460	16	0	0	96	230	326	11
2048	0	134	134	5	0	0	0	0	0	0
2049	0	134	134	5	0	0	134	0	134	5
2050	0	0	0	0	0	0	0	0	0	0

Note: The number of canisters listed are for storage of irradiated fuel not GTCC waste.

Table 4A
SONGS Unit 2
Decommissioning Funding Plan

Year	Radiological Decontamination	Spent Fuel Management	Site Restoration	Total Decommissioning Costs	Available Funds
2013	$25,749	$63,891	$49,067	$138,706	$1,847,000
2014	$79,799	$35,719	$15,089	$130,607	
2015	$69,196	$106,308	$7,439	$182,943	
2016	$54,541	$59,308	$3,730	$117,579	
2017	$111,903	$59,308	$1,957	$173,168	
2018	$47,520	$59,308	$0	$106,828	
2019	$108,328	$27,554	$13,539	$149,420	
2020	$185,482	$4,908	$36	$190,426	
2021	$79,081	$4,908	$36	$84,026	
2022	$54,785	$4,908	$1,927	$61,621	
2023	$158,207	$4,908	$36	$163,151	
2024	$37,930	$4,908	$16,848	$59.687	
2025	$2,922	$4,908	$44,621	$52,451	
2026	$2,922	$4,908	$19,412	$27,243	
2027	$2,922	$4,908	$22,469	$30,299	
2028	$2,922	$4,908	$31,688	$39,518	
2029	$2,922	$4,908	$66,873	$74,704	
2030	$2,922	$4,908	$71,867	$79,697	
2031	$2,055	$5,089	$23,181	$30.325	
2032	$2,122	$7,214	$0	$9,336	
2033	$0	$7,214	$0	$7,214	
2034	$0	$7,214	$0	$7,214	
2035	$0	$7,228	$0	$7,228	
2036	$0	$7,665	$0	$7,665	
2037	$0	$7,665	$0	$7,665	
2038	$0	$7,665	$0	$7,665	
2039	$0	$7,665	$0	$7,665	
2040	$0	$7,665	$0	$7,665	
2041	$0	$7,665	$0	$7,665	
2042	$0	$7,665	$0	$7,665	
2043	$0	$7,665	$0	$7,665	
2044	$0	$7,665	$0	$7,665	
2045	$0	$7,665	$0	$7,665	
2046	$0	$7,665	$0	$7,665	
2047	$0	$7,665	$0	$7,665	
2048	$0	$7,665	$0	$7,665	
2049	$0	$7,667	$0	$7,667	
2050	$0	$9,974	$20,177	$30,151	
2051	$0	$6,573	$11,928	$18,500	
2052	$0	$0	$1,377	$1,377	

Notes: Costs are in 2014 dollars (in thousands) and are not escalated from the base year
SONGS Unit 2 Trust fund balances at end of 2013 were $1,847,000

Table 4B
SONGS Unit 3
Decommissioning Funding Plan

Year	Radiological Decontamination	Spent Fuel Management	Site Restoration	Total Decommissioning Costs	Available Funds
2013	$26,566	$66,105	$49,067	$141,739	$2,079,400
2014	$78,964	$40,156	$15,969	$135,089	
2015	$74,096	$112,024	$9,390	$195,509	
2016	$61,451	$64,405	$25,227	$151,083	
2017	$40,631	$64,405	$3,799	$108,835	
2018	$86,348	$64,405	$0	$150,753	
2019	$96,521	$29,675	$13,908	$140.014	
2020	$120,873	$4,908	$2,135	$127,916	
2021	$194,090	$4,908	$575	$199,574	
2022	$135,313	$4,908	$2,467	$142,688	
2023	$114,581	$4,908	$1,511	$121,000	
2024	$26,874	$4,908	$36,778	$68,560	
2025	$2,922	$4,908	$40,655	$48,485	
2026	$2,922	$4,908	$21,676	$29,507	
2027	$2,922	$4,908	$25,848	$33,678	
2028	$2,922	$4,908	$20,945	$28,776	
2029	$2,922	$4,908	$117,321	$125,151	
2030	$2,922	$4,908	$116,672	$124,503	
2031	$2,055	$5,089	$25,501	$32,645	
2032	$2,122	$7,214	$0	$9,336	
2033	$0	$7,214	$0	$7,214	
2034	$0	$7,214	$0	$7,214	
2035	$0	$7,228	$0	$7,228	
2036	$0	$7,665	$0	$7,665	
2037	$0	$7,665	$0	$7,665	
2038	$0	$7,665	$0	$7,665	
2039	$0	$7,665	$0	$7,665	
2040	$0	$7,665	$0	$7,665	
2041	$0	$7,665	$0	$7,665	
2042	$0	$7,665	$0	$7,665	
2043	$0	$7,665	$0	$7,665	
2044	$0	$7,665	$0	$7,665	
2045	$0	$7,665	$0	$7,665	
2046	$0	$7,665	$0	$7,665	
2047	$0	$7,665	$0	$7,665	
2048	$0	$7,665	$0	$7,665	
2049	$0	$7,667	$0	$7,667	
2050	$0	$9,974	$23,120	$33,094	
2051	$0	$6,573	$45,566	$52,139	
2052	$0	$0	$1,377	$1,377	

Notes: Costs are in 2014 dollars (in thousands) and are not escalated from the base year
SONGS Unit 3 Trust Fund balances at end of 2013 were $2,079,400

SOUTHERN CALIFORNIA
EDISON®
An EDISON INTERNATIONAL® Company

Thomas J. Palmisano
Vice President & Chief Nuclear Officer

10 CFR 50.82(a)(4)(i)

September 23, 2014

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington D.C. 20555-0001

Subject: **Docket Nos. 50-361 and 50-362,**
San Onofre Nuclear Generating Station, Units 2 and 3
Site Specific Decommissioning Cost Estimate

References:

1. Letter from P. T. Dietrich (SCE) to the U.S. Nuclear Regulatory Commission dated June 12, 2013; Subject: Certification of Permanent Cessation of Power Operations San Onofre Nuclear Generating Station, Units 2 and 3
2. Letter from Thomas J. Palmisano (SCE) to the U.S. Nuclear Regulatory Commission dated February 13, 2014; Subject: Access to Nuclear Decommissioning Trust Funds, San Onofre Nuclear Station, Units 2 and 3
3. Letter from Richard C. Brabec (SCE) to the U.S. Nuclear Regulatory Commission dated March 12, 2014; Subject: Access to Decommissioning Trust Funds, San Onofre Nuclear Generating Station Units 2 and 3
4. Letter from Richard C. Brabec (SCE) to the U.S. Nuclear Regulatory Commission dated March 31, 2014; Subject: 10 CFR 50.75(f)(1) Decommissioning Funding Status Report, San Onofre Nuclear Generating Station Units 2 and 3

Dear Sir or Madam:

On June 12, 2013, in accordance with 10 CFR 50.82(a)(1)(i), Southern California Edison (SCE) submitted a letter to the U.S. Nuclear Regulatory Commission (NRC) (Reference 1) certifying the permanent cessation of operations at San Onofre Nuclear Generating Station (SONGS), Units 2 and 3. In accordance with 10 CFR 50.54(bb) and 10 CFR 50.82(a)(4)(i), SCE is required to submit an Irradiated Fuel Management Plan (IFMP), Site Specific Decommissioning Cost Estimate (DCE) and Post-Shutdown Decommissioning Activities Report (PSDAR) within two years of permanent cessation of operations.

The SONGS, Units 2 and 3 DCE is attached. The SONGS, Units 2 and 3 IFMP and PSDAR are being concurrently submitted under separate cover letters. The DCE provides more current estimates of annual cash flow than were previously provided in the Nuclear Decommissioning Trust Fund Exemption Request (References 2 and 3) and annual funding assurance update (Reference 4). Future filings with the California Public Utilities Commission will be based on the SONGS, Units 2 and 3 DCE and subsequent revisions.

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A001
NRR

The descriptions of decommissioning activities and phases in the DCE are consistent with those described in the PSDAR. Both the DCE and PSDAR represent SCE's current plans and are subject to change as the project progresses. Much of the third-party contracting activities associated with decommissioning are underway but have not been finalized. As contracts are finalized and SCE progresses through the actual work of the decommissioning project, various risks will be realized or avoided and contingencies adjusted, accordingly.

Changes to significant details will be included in subsequent revisions to the DCE as required by 10 CFR 50.54(bb). Financial assurance information will be provided on an annual basis as required by 10 CFR 50.75(f)(1).

This letter does not contain any new commitments.

If there are any questions or if additional information is needed, please contact me or Ms. Andrea Sterdis at (949) 368-9985.

Sincerely,

Enclosure: San Onofre Nuclear Generating Station Units 2 and 3 Site Specific Decommissioning Cost Estimate

cc: M. L. Dapas, Regional Administrator, NRC Region IV
T. J. Wengert, NRC Project Manager, San Onofre Units 2 and 3 Decommissioning
R. E. Lantz, NRC Region IV, San Onofre Units 2 and 3
G. G. Warrick, NRC Senior Resident Inspector, San Onofre Units 2 and 3
S. Y. Hsu, California Department of Health Services, Radiologic Health Branch

ENERGY*SOLUTIONS* **Document No. 164001-DCE-001**

2014 Decommissioning Cost Analysis of the San Onofre Nuclear Generating Station Units 2 & 3

Project No. 164001 **Rev 1**

Prepared for:
Southern California Edison.
2244 Walnut Grove Avenue
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Prepared by:
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Authored By:	Michael S. Williams Michael S. Williams, Project Manager	September 5, 2014 Date
Reviewed By:	Barry Sims Barry S. Sims, Technical Advisor	September 5, 2014 Date
Approved By	Michael S. Williams Michael S. Williams, Project Manager	September 5, 2014 Date

[ ] New Report
[ ] Title Change
[x] Report Revision
[ ] Report Rewrite

Effective Date Sept 5, 2014

SONGS UNIT-2 AND UNIT-3
DECOMMISSIONING COST ESTIMATE
DESCRIPTION OF REVISION

MAJOR REVISION ______ MINOR REVISION_X____
REVISION NUMBER – 1 EFFECTIVE DATE - 9/5/2014

The revisions contained in this MINOR REVISION to the SONGS Unit-2 and Unit-3 Decommissioning Cost Estimate are minor in nature and do not revise or otherwise impact the content or results of the cost estimate.

ITEM-1

A new Appendix-F is added to the DCE at the request of San Diego Gas & Electric Company (SDG&E) in order to provide information regarding its internal decommissioning costs which it expects to incur and to fund on its own behalf in addition to its 20% share of the Decommissioning Cost Estimate.

ITEM-2

The APPENDICES section of the DCE Table of Contents is revised to include the new APPENDIX-F SDG&E SONGS Decommissioning Costs (100%)

ITEM-3

Within the narrative section of the DCE the various appearances of the term "utility staff" have been revised to include a parenthetic statement "(Licensee)" to clarify that the utility staff means the NRC Licensee.

ITEM-4

On Table 6-1 "Cost and Schedule Summary" the title block for SPENT FUEL is revised to include "(72.30)" since this section also contains cost elements associated with ISFSI decommissioning.

ITEM-5

Added new SDG&E footnote for Table 1-1 referring to Appendix F

TABLE OF CONTENTS

FIGURES

TABLES

APPENDICES

ACRONYMS AND ABBREVIATIONS

AHSM	Advanced Horizontal Storage Modules
AIF	Atomic Industrial Forum
ALARA	As Low As Reasonably Achievable
ARO	Asset Retirement Obligation
CFR	Code of Federal Regulations
CPM	Critical Path Method
DAW	Dry Active Waste
DGC	Decommissioning General Contractor
DOE	U.S. Department of Energy
DSC	Dry Shielded Canister
ESS	Essential System
FEMA	Federal Emergency Management Agency
FSS	Final Status Survey
FTE	Full Time Equivalent
GSA	U.S. General Services Administration
GTCC	Greater Than Class C
HP	Health Physics
ISFSI	Independent Spent Fuel Storage Installation
LLRW	Low-Level Radioactive Waste
LLW	Low Level Waste
LLWPA	Low-Level Waste Policy Act
LOP	Life-of-Plant
MARSSIM	Multi-Agency Radiation Survey and Site Investigation Manual
MPC	Multi-Purpose Canister
MWt	Megawatt thermal
NON	Non-Essential System
NRC	Nuclear Regulatory Commission
NSSS	Nuclear Steam Supply System
ORISE	Oak Ridge Institute for Science and Education
PCB	Polychlorinated Biphenyl
PGE	Pacific Gas & Electric
PSDAR	Post-Shutdown Decommissioning Activities Report
PWR	Pressurized Water Reactor
RIF	Reduction In Force
SCE	Southern California Edison
SONGS	San Onofre Nuclear Generating Station
STRUCT	Structure
TCEQ	Texas Commission on Environmental Quality
WBS	Work Breakdown Structure
WCS	Waste Control Specialists LLC
UCF	Unit Cost Factor

1.0 EXECUTIVE SUMMARY

This report presents the 2014 Decommissioning Cost Estimate (DCE) Study of the San Onofre Nuclear Generating Station (SONGS) Units 2 & 3, hereinafter referred to as the 2014 Cost Study. The San Onofre Nuclear Generating Station is operated by the Southern California Edison Company (SCE).

On June 7, 2013, SCE announced its intention to permanently cease power generation operations and shut down SONGS Units 2 & 3. Units 2 & 3 had not produced power since January 9, 2012 and January 31, 2012, respectively. SCE now has the responsibility to decommission the site. In January 2014 SCE contracted with Energy*Solutions* to evaluate decommissioning alternatives and assist in the development of a detailed project schedule and DCE to support the preparation and submittal of a Post Shutdown Decommissioning Activities Report (PSDAR) in accordance with 10 CFR 50.82(a)(4)(i), which requires that a PSDAR be submitted within two years following the permanent cessation of operations.

This study has been performed to furnish an estimate of the costs for: (1) decommissioning SONGS Units 2 & 3 to the extent required to terminate the plant's operating license pursuant to 10 CFR 50.75(c); (2) post-shutdown management of spent fuel until acceptance by the U.S. Department of Energy (DOE) pursuant to 10 CFR 50.54(bb); (3) demolition of uncontaminated structures and restoration of the site in accordance with the United States Department of Navy Grant of Easement (Ref. No. 14); and the California State Lands Commission Easement Lease (Ref. No. 15); and (4) Independent Spent Fuel Storage Installation (ISFSI) decommissioning pursuant to 10 CFR 72.30. This study includes SCE's actual costs incurred in the transitional periods following cessation of permanent operations on June 7, 2013 until December 31, 2013. Costs presented herein commencing on January 1, 2014 are estimated.

SCE's December 2012 testimony to the CPUC provided the basis for the current spent fuel management costs. SCE is continuing to review available information from the DOE to determine if the DOE start date assumption of 2024 requires updating. The DCE will be revised accordingly as new information becomes available.

Accordingly, the costs and schedules for all activities are segregated for regulatory purposes as follows: costs for "License Termination" (10 CFR 50.75(c)); costs for "Spent Fuel Management" (10 CFR 50.54(bb)); costs for "Site Restoration" (clean removal and site restoration) final site conditions; and costs for "ISFSI Decommissioning" (10 CFR 72.30). Energy*Solutions* has established a Work Breakdown Structure (WBS) and cost accounting system to differentiate between these project accounts.

This study analyzes the following technical approach to decommissioning as defined by SCE:

- DECON methodology.
- Permanent cessation of operations on June 7, 2013.
- Termination of spent fuel pool operation six years after permanent shutdown.
- Spent fuel will be stored in Multi-Purpose Canisters (MPCs) at an on-site Independent Spent Fuel Storage Installation (ISFSI).

- A dry transfer facility will not be necessary for transfer of SNF canisters for transport.
- DOE begins accepting spent fuel from the industry in 2024 and completes the removal of all SONGS spent fuel by 2049.
- Decommissioning will be performed by a Decommissioning General Contractor (DGC) with oversight by the SONGS participants.
- Incorporation of Life-of-Plant (LOP) Disposal Rates for Class A Low-Level Radioactive Waste (LLRW).
- Incorporation of disposal rates for Class B and C LLRW based on recent quotes for disposal at the Waste Control Specialists LLC (WCS) site in Andrews County, Texas.

The cost estimate results are provided in Table 1-1. Table 1-1 gives License Termination costs (which correspond to 10 CFR 50.75 (c) requirements); Spent Fuel Management costs (which correspond to 10 CFR 50.54 (bb) requirements); and Site Restoration costs (which correspond to activities such as clean building demolition and site grading and end-state preparation as required under the Site Easement).

Table 1-1
Decommissioning Cost Summary[1][2]
(2014 Dollars in Thousands)

Cost Account	Unit 2	Unit 3	Total
License Termination 50.75(c)	$1,034,230	$1,078,016	**$2,112,246**
Spent Fuel Management 50.54(bb)	$623,209	$652,987	**$1,276,196**
Site Restoration	$423,297	$599,507	**$1,022,804**
Totals	**$2,080,735**	**$2,330,511**	**$4,411,246**

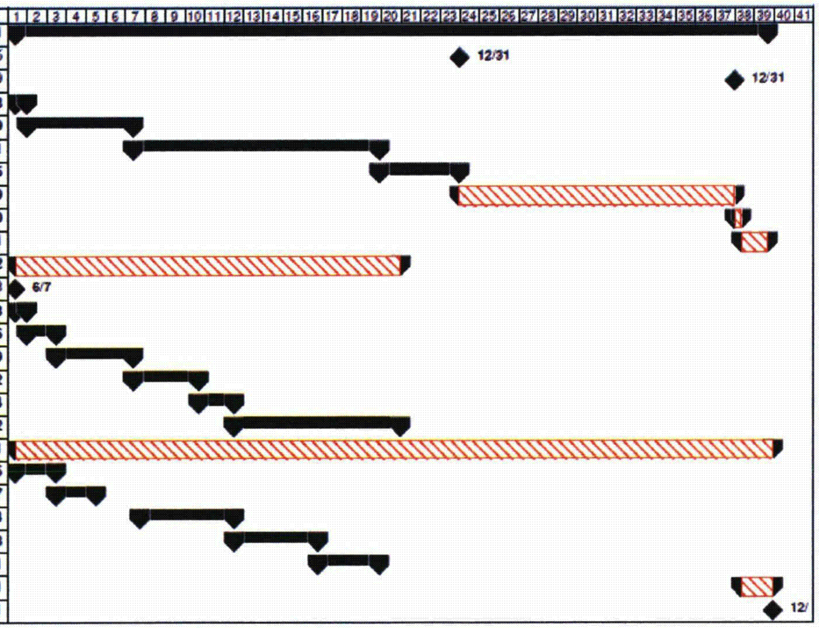
The estimate is based on site-specific plant systems and buildings inventories. These inventories, and Energy*Solutions*' proprietary Unit Cost Factors (UCFs), were used to generate required manhours, activity schedule hours and costs, and waste volume, weight, and classification. Based on the activity schedule hours and a decommissioning activities analysis, a Critical Path Method (CPM) analysis was performed to determine the decommissioning schedules. These schedules reflect the effects of sequenced activity-dependent or distributed decommissioning elements such as planning and preparations, major component removal, building decontamination, and spent fuel shipping. The schedules are divided into project phases (periods) and presented, as noted previously, by cost account "License Termination," "Spent Fuel Management," or "Site Restoration." The summary is shown in Figure 1-1, and may also be found in Section 6.0 of this report.

[1] In addition, the Decommissioning Cost Summary in Table 1-1 does not include separate internal costs that San Diego Gas & Electric Company (SDG&E) has indicated that it expects to incur. SDG&E provides information regarding these costs in Appendix F
[2] Rows and columns may not add correctly due to rounding.

Figure 1-1
Summary Schedule

DECON with Dry Storage, 2013 Shutdown and DOE Acceptance in 2024

Task Name	Start	Finish
Post-Shutdown Spent Fuel Management	06/07/2013	09/08/2051
Spent Fuel Shipping Complete - Unit 1	12/31/2035	12/31/2035
Spent Fuel Shipping Complete - Units 2 & 3	12/31/2049	12/31/2049
SNF Pd 1 - Spent Fuel Management Transition	06/07/2013	12/31/2013
SNF Pd 2 - Spent Fuel Transfer to Dry Storage	1/1/2014	06/01/2019
SNF Pd 3 - Dry Storage During Decommissioning - Units 1, 2 and 3	06/01/2019	12/05/2031
SNF Pd 4 - Dry Storage Only - Units 1, 2 and 3	12/05/2031	12/31/2035
SNF Pd 5 - Dry Storage Only - Units 2 and 3	12/31/2035	12/31/2049
SNF D&D Pd 1 - ISFSI D&D Planning	12/31/2049	05/06/2050
SNF D&D Pd 2 - ISFSI D&D	05/06/2050	09/08/2051
Part 50 License Termination	06/07/2013	12/24/2032
Announcement of Cessation of Operations (June 7, 2013)	06/07/2013	06/07/2013
Decon Pd 1 - Transition to Decommissioning	06/07/2013	12/31/2013
Decon Pd 2 - Decommissioning Planning and Site Modifications	1/1/2014	06/30/2015
Decon Pd 3 - Decommissioning Preparations and Reactor Internals Segmentation	06/30/2015	06/01/2019
Decon Pd 4 - Plant Systems and Large Component Removal	06/01/2019	09/24/2022
Decon Pd 5 - Building Decontamination	09/24/2022	07/13/2024
Decon Pd 6 - License Termination During Demolition	07/13/2024	12/24/2032
Site Restoration	06/07/2013	12/15/2051
SR Pd 1 - Transition to Site Restoration	06/07/2013	06/30/2015
SR Pd 2 - Building Demolition During Decommissioning	06/30/2015	07/11/2017
SR Pd 3 - Subsurface Demolition Engineering and Permitting	10/01/2019	07/13/2024
SR Pd 4 - Building Demolition to 3 Feet Below Grade	07/13/2024	10/14/2028
SR Pd 5 - Subgrade Structure Removal Below -3 Feet	10/14/2028	12/05/2031
SR Pd 6 - Final Site Restoration and Lease Termination	05/06/2050	12/15/2051
Final Easement Termination	12/15/2051	12/15/2051



2.0 INTRODUCTION

2.1 Study Objective

This report presents the 2014 Decommissioning Cost Estimate Study of the San Onofre Nuclear Generating Station (SONGS) Units 2 & 3, hereinafter referred to as the 2014 Cost Study. The San Onofre Nuclear Generating Station is owned by the Southern California Edison Company (SCE), San Diego Gas & Electric Company, and the City of Riverside. A former owner, the City of Anaheim, also has liability for decommissioning. SCE has provided the following information regarding the liability by owner for SONGS decommissioning costs:

Cost Categories	**Owners**			
	SDG&E	**Riverside**	**Anaheim**	**SCE**
SONGS 1	20%	0%	0%	80%
SONGS 2	20%	1.79%	2.4737%	75.7363%
SONGS 3	20%	1.79%	2.4625%	75.7475%
Common Facilities (Units 2 & 3)	20%	1.79%	2.4681%	75.7419%
SONGS 1 Fuel	20%	0%	0%	80%
SONGS 2/3 Fuel	20%	1.79%	2.3398%	75.8702%
ISFSI Maintenance and D&D	20%	1.6066%	2.2686%	76.1248%
San Diego Switchyard	100%	0%	0%	0%
Edison Switchyard	0%	0%	0%	100%
Interconnection Facilities	50%	0%	0%	50%
Nuclear Fuel Cancellation Charges	20%	1.79%	0%	78.21%

This study has been performed to support the development of a site-specific PSDAR and furnish an estimate of the costs for (1) decommissioning SONGS Units 2 & 3 to the extent required to terminate the plant's operating license, (2) post-shutdown management of spent fuel until acceptance by the U.S. Department of Energy (DOE), (3) demolition of uncontaminated structures and restoration of the site in accordance with the U.S. Department of Navy Grant of Easement (Ref. No. 14), and the California State Lands Commission Easement Lease (Ref. No. 15), and (4) Independent Spent Fuel Storage Installation (ISFSI) decommissioning. This study also includes SCE's actual costs incurred in the transitional periods following cessation of permanent operations until December 31, 2013. Estimated costs begin on January 1, 2014.

The study methodology follows the basic approach originally presented in the Atomic Industrial Forum/National Environmental Studies Project Report AIF/NESP-036, "Guidelines for Producing Commercial Nuclear Power Plant Decommissioning Cost Estimates," (Ref. No. 2). The report was prepared in accordance with Nuclear Regulatory Commission (NRC) Regulatory Guide 1.202, "Standard Format and Content of Decommissioning Cost Estimates for Nuclear Power Reactors," (Ref. No. 3). The estimate is based on compliance with current regulatory requirements and proven decommissioning technologies.

NRC requirements, set forth in Title 10 of the Code of Federal Regulations (CFR), differentiate between the post-shutdown costs associated with the decommissioning of the nuclear plant facility, those associated with storage of spent fuel on-site, and those associated with the decommissioning of the spent fuel storage facility. The Code of Federal Regulations, however, does not address the entire scope of the decommissioning liability for each nuclear facility. 10 CFR 50.75(c) requires funding by the licensee(s) of the facility for the decommissioning program, but specifically excludes the cost of removal and disposal of spent fuel and structures that do not require disposal as radioactive material. 10 CFR 50.75(c) also excludes the cost of site restoration activities that do not involve the removal of residual radioactivity necessary to terminate the NRC license(s). 10 CFR 50.54 (bb) requires funding by the licensee(s) "for the management of all irradiated fuel at the reactor upon expiration of the reactor operating license(s) until title to the irradiated fuel and possession of the fuel is transferred to the Secretary of Energy for its ultimate disposal in a repository." 10 CFR 72.30 requires funding for decommissioning of the on-site spent fuel storage facility after the irradiated fuel is accepted by the DOE.

In addition to the NRC Decommissioning requirements described above, the Site Easements require the demolition and removal of all improvements installed on both the on-shore and off-shore sites, including all substructures regardless of depth, and site restoration to the satisfaction of the Grantors.

This study analyzes the following technical approach to decommissioning as defined by SCE and the co-owners:

- DECON methodology.
- Permanent cessation of operations and commencement of decommissioning planning on June 7, 2013.
- Termination of spent fuel pool operation within six years after permanent shutdown.
- Spent fuel will be stored in transportable Multi-Purpose Canisters (MPCs) at an on-site Independent Spent Fuel Storage Installation (ISFSI).
- A dry transfer facility will not be necessary for transfer of SNF canisters for transport.
- DOE begins accepting spent fuel from the industry in 2024 and completes the removal of all SONGS spent fuel by 2049.
- Decommissioning will be performed by a Decommissioning General Contractor (DGC) with oversight by the SONGS participants.

In addition, this study includes the following assumptions:

- Incorporation of Energy*Solutions*' Life-of-Plant (LOP) Disposal Rates for Class A Low-Level Radioactive Waste (LLRW), (Ref. No. 7).
- Incorporation of disposal rates for Class B and C LLRW based on recent quotes for disposal at the Waste Control Specialists LLC (WCS) site in Andrews County, Texas.

2.2 Regulatory Framework

Provisions of current laws and regulations affecting decommissioning, waste management, and spent fuel management are as follows:

1. NRC regulations require a license for on-site storage of spent fuel. Wet storage in a spent fuel pool is authorized by a facility's 10 CFR Part 50 license. On-site dry storage of spent fuel at an Independent Spent Fuel Storage Installation (ISFSI) is licensed by either: (a) the general license set forth in 10 CFR 72.210, which requires that a Part 50 license be in place; or (b) a site-specific ISFSI license issued pursuant to 10 CFR Part 72.

2. 10 CFR 50.75(c) requires funding by the licensee(s) of the facility for decommissioning.

3. 10 CFR 50.54 (bb) requires the licensee(s), within two years following permanent cessation of operation of the reactor or five years before expiration of the operating license(s), whichever occurs first, to submit written notification to the NRC for its review and preliminary approval of the program by which the licensee intends to manage and provide funding "for the management of all irradiated fuel at the reactor upon expiration of the reactor operating license until title to the irradiated fuel and possession of the fuel is transferred to the Secretary of Energy for its ultimate disposal in a repository."

4. 10 CFR 961 (Ref. No. 4), Appendix E, requires spent fuel to be cooled for at least five years before it can be accepted by DOE as "standard spent fuel."

5. 10 CFR 72.30 requires funding by the licensee(s) for termination of the ISFSI license.

<u>Decommissioning Alternatives</u>

The three basic methods for decommissioning are DECON, SAFSTOR, and ENTOMB, which are summarized as follows:

1. DECON: The equipment, structures, and portions of the facility and site that contain radioactive contaminants are promptly removed or decontaminated to a level that permits termination of the license after cessation of operations.

2. SAFSTOR: The facility is placed in a safe, stable condition and maintained in that state (safe storage). The facility is decontaminated and dismantled at the end of the storage period to levels that permit license termination. NRC regulations require decommissioning to be completed within 60 years of cessation of operation.

3. ENTOMB: Radioactive structures, systems, and components are encased in a structurally long-lived substance, such as concrete. The entombed structure is appropriately maintained and monitored until radioactivity decays to a level that permits termination of the license. Since entombment will exceed the requirement

for decommissioning to be completed within 60 years of cessation of operation, NRC handles entombment requests on a case-by-case basis.

Post-Shutdown Spent Fuel Management Alternatives

The options for long-term post-shutdown spent fuel management currently available to power plant operators are (1) wet storage consisting of continued maintenance and operation of the spent fuel pool, and (2) dry storage consisting of transfer of spent fuel from the fuel pool to on-site dry storage modules after a cooling period or any combination of the two as is the present case at SONGS. Maintaining the spent fuel pool for an extended duration following cessation of operations prevents termination of the Part 50 license and typically has a higher annual maintenance and operating cost than the dry storage alternative. Transfer of spent fuel to an ISFSI requires additional expenditures for purchase and construction of the ISFSI and dismantlement and disposal of the ISFSI following completion of spent fuel transfer to DOE.

The spent fuel shipping schedules furnished by SCE for this study are based on projections that DOE will commence accepting spent fuel from domestic commercial nuclear power plants in 2024, and that the DOE will accept spent fuel at the rate published in DOE's July 2004 Acceptance Priority Ranking & Annual Capacity Report (DOE/RW-0567) (Ref. No. 12). These assumptions are in accordance with SCE testimony to the Public Utilities Commission of the State of California (Ref. No. 17). Additionally, SCE is reviewing available information from the DOE to determine if the DOE start date assumption requires updating. The DCE will be revised accordingly as new information becomes available.

3.0 STUDY METHODOLOGY

3.1 General Description

Energy*Solutions* maintains a proprietary decommissioning cost model based upon the fundamental technical approach established in AIF/NESP-036, "Guidelines for Producing Commercial Nuclear Power Plant Decommissioning Cost Estimates," dated May 1986 (Ref. No. 2). The cost model has been updated frequently in accordance with regulatory requirements and industry experience. The cost model includes elements for estimating distributed and undistributed costs. Distributed costs are activity specific and include planning and preparation costs as well as costs for decontamination, packaging, disposal, and removal of major components and systems. For example, costs for the segmentation, packaging, and disposal of the reactor internals are distributed costs. Undistributed costs, sometimes referred to as collateral costs, are typically time dependent costs such as utility (Licensee) and decommissioning general contractor staff, property taxes, insurance, regulatory fees and permits, energy costs, and security staff.

The methodology for preparing cost estimates for a selected decommissioning alternative requires development of a site-specific detailed work activity sequence based upon the plant inventory. The activity sequence is used to define the labor, material, equipment, energy resources, and duration required for each activity. In the case of major components, individual work sequence activity analyses are performed based on the physical and radiological characteristics of the component, and the packaging, transportation, and disposal options available.

In the case of structures and small components and equipment such as piping, pumps, and tanks, the work durations and costs are calculated based on UCFs. UCFs are economic parameters developed to express costs per unit of work output, piece of equipment, or time. They are developed using decommissioning experience, information on the latest technology applicable to decommissioning, and engineering judgment. The total cost of a specific decommissioning activity can be determined by multiplying the total number of units associated with that activity by the UCF, expressed as $/unit, for that activity. For example, the estimated demolition cost of a non-contaminated concrete structure can be obtained by multiplying the volume of concrete in the structure by the UCF for non-contaminated reinforced concrete demolition, expressed in $/unit volume. Each UCF has associated with it a man-hours/unit and schedule-hours/unit. From these values, total man-hours and total schedule-hours can be estimated for a particular activity.

3.2 Schedule Analysis

After the work activity durations are calculated for all distributed activities, a critical path schedule analysis is performed using MS Project. The schedule accounts for constraints such as spent fuel cooling periods and regulatory reviews. The schedule is typically delineated into phases or time periods (hereinafter referred to as period or periods) that differentiate manpower requirements and undistributed costs.

In order to differentiate between License Termination, Spent Fuel, and Site Restoration elements of the entire decommissioning scope of work, Energy*Solutions* has established a Work Breakdown Structure (WBS) and cost accounting system to treat each element as a subproject.

Accordingly, the overall project schedule is divided into interrelated periods with major milestones defining the beginning and ending of each period. The major milestones also serve as the basis for integrating the periods of the three subprojects.

3.3 Decommissioning Staff

Energy*Solutions* has assumed that the SONGS Units 2 and 3 decommissioning project will be performed in an efficiently planned and executed manner using project personnel experienced in decommissioning. This DCE assumes that the decommissioning will be performed by a highly experienced and qualified DGC, with oversight and management of the decommissioning operations performed by the Licensee staff. It is also assumed that the Utility (Licensee) staff will be supplemented by a professional consulting engineering firm, particularly in the planning and preparation phase.

Energy*Solutions* analyzed the SONGS licensee staff and developed a site-specific staffing plan. The SCE existing salary structure was then used as the basis for calculating Utility (Licensee) staff labor costs. Energy*Solutions* used industry data to develop DGC salary costs.

Staffing levels, for both staffing plans and for each project period, are based on the Atomic Industrial Forum (AIF) guidelines and industry experience. The sizes of the staffs are varied in each period in accordance with the requirements of the work activities. Staffing has been organized into the following departments or functional groups:

- Decommissioning
- Engineering
- Maintenance and Work Control
- Operations
- Oversight and Nuclear Safety
- Radiation Protection and Chemistry
- Regulatory and Emergency Planning
- Safety and Human Performance
- Security Administration
- Security Guard Force
- Site Management and Administration
- Additional Staff for Spent Fuel Shipping
- DGC Staff

3.4 Waste Disposal

Waste management costs comprise a significant portion of the decommissioning cost estimate. Additionally, limited future access to disposal sites licensed for receipt of Class B and C wastes introduces a significant level of uncertainty with respect to the appropriateness of using existing rate structures to estimate disposal costs of these wastes. Energy*Solutions*' approach to estimating waste disposal costs is discussed in the following paragraphs.

<u>Waste Classification</u>

Regulations governing disposal of radioactive waste are stringent in order to ensure control of the waste and preclude adverse impact on public health and safety. At present, LLRW disposal

is controlled by 10 CFR 61, which went into effect in December 1983. This regulation stipulates the criteria for the establishment and operation of shallow-land LLRW burial facilities. Embodied within this new regulation are criteria and classifications for packaging LLRW such that it is acceptable for burial at licensed LLRW disposal sites.

For each waste classification, 10 CFR 61 stipulates specific criteria for physical and chemical properties that the LLRW must meet in order to be accepted at a licensed disposal site. The LLRW disposal criteria of 10 CFR 61 require that LLRW generators determine the proportional amount of a number of specific radioactive isotopes present in each container of disposable LLRW. This requirement for isotopic analysis of each container of disposable LLRW is met by employing a combination of analytical techniques such as computerized analyses based upon scaling factors, sample laboratory analyses, and direct assay methods. Having performed an isotopic analysis of each container of disposable LLRW, the waste must then be classified according to one of the classifications (Class A, B, C, or Greater Than Class C (GTCC)) as defined in 10 CFR 61.

Energy*Solutions*' classification of LLRW resulting from decommissioning activities is based on AIF/NESP-036 (Ref. No. 2), NUREG/CR-0130 (Ref. No. 5), NUREG/CR-0672 (Ref. No. 6), and recent industry experience. The estimated curie content of the reactor vessel and internals at shutdown is derived from NUREG/CR-0130 for Pressurized Water Reactors (PWRs) and NUREG/CR-0672 for Boiling Water Reactors (BWRs), and adjusted for the different mass of components and period of decay.

<u>Packaging</u>

Selection of the type and quantity of containers required for Class B and C wastes is based on the most restrictive of either curie content, dose-rate, container weight limit, or container volume limit. GTCC wastes from segmentation of the reactor vessel internals is packaged in spent fuel canisters. The selection of container type for Class A waste is based on the transportation mode (rail, truck, barge, etc.) and waste form. The quantity of Class A waste containers is determined by the most restrictive of either container weight limit or container volume limit. Large components, such as steam generators, pressurizers, and reactor recirculation pumps, are shipped as their own containers with additional shielding as required.

Container costs are obtained from manufacturers specializing in the design and fabrication of storage containers for nuclear materials. Shielded transport cask and liner costs are obtained from the cask owners and operators.

<u>Transportation</u>

Transportation routes to processing and disposal facilities are determined based on available transportation modes (truck, rail, barge, or combinations). Transportation costs for the selected routes and modes are obtained from vendor quotes or published tariffs whenever possible.

<u>Class A Disposal Options and Rates</u>

In accordance with the existing Life-of-Plant Disposal Agreement (Ref. No. 7), all Class A waste that meets the waste acceptance criteria are to be disposed of at Energy*Solutions*' LLRW

disposal facility in Clive, Utah. All reported waste disposal costs include packaging, transportation, and any applicable surcharges.

Class B and C Disposal Options and Rates

Currently, within the United States, there are only three operational commercial near-surface disposal facilities licensed to accept Class B and C LLRW: the Barnwell facility, operated by Energy*Solutions* in Barnwell, South Carolina; the U.S. Ecology facility in Richland, Washington; and the recently licensed facility in Andrews County, Texas operated by Waste Control Specialists. Barnwell only accepts waste from states within the Atlantic Compact and U.S. Ecology only accepts waste from states within the Northwest and Rocky Mountain Compacts. However, the WCS facility will accept waste from the Texas Compact (comprised of Texas and Vermont) and from non-Compact generators. The Texas Compact Commission on March 23, 2012 approved amendments to rules allowing the import of non-compact generator LLRW for disposal at the WCS Andrews County facility.

Greater Than Class C (GTCC)

Wastes identified as 10 CFR 61 Class A, B, and C may be disposed of at near-surface disposal facilities. Certain components are highly activated and may exceed the radionuclide concentration limitations for 10 CFR 61 Class C waste. In accordance with 10 CFR 61, these components, which are referred to as Greater Than Class C (GTCC) wastes, cannot be disposed of in a near-surface LLRW disposal facility and must be transferred to a geologic repository or a similar site approved by the NRC.

Highly activated sections of the reactor vessel internals will result in GTCC waste. Presently, a facility does not exist for the disposal of wastes exceeding 10 CFR 61 Class C limitations. Energy*Solutions* assumes that the DOE will accept this waste along with spent fuel. Although courts have held that DOE is obligated to accept and dispose of GTCC, issues regarding potential costs remain potentially unsettled. Therefore, Energy*Solutions* conservatively estimates a GTCC waste disposal cost. Energy*Solutions* assumes that the GTCC waste will be packaged in spent fuel canisters and will be shipped to a storage or disposal facility operated by DOE along with the spent fuel. Additionally, Energy*Solutions* assumes shipping costs for GTCC waste to be equivalent to the commercial cost of shipping a Type B licensed, shielded cask such as the CNS 8-120B cask, which is owned and operated by Energy*Solutions*.

LLRW Volume Reduction

Becasue current Class A LLRW disposal rates are significantly lower than LLRW volume reduction rates, Energy*Solutions* does not assume on-site volume reduction techniques such as waste compaction or an aggressive decontamination, survey and release effort.

Non-Radioactive Non-Hazardous Waste Disposal

Energy*Solutions* assumes that recyclable, non-radioactive scrap metal resulting from the decommissioning program will be sold to a scrap metal dealer. However, no cost credit is assumed in the estimate for the value of the scrap metal. Clean (non-contaminated) concrete and demolition debris is assumed to be removed off site to an out of state Class III landfill consistent

with the Governor of the State of California Executive Order D-62-02 (Ref. No. 16). This study includes the costs of installation and operation of Energy*Solutions*' GAmma Radiation Detection and In-container ANalysis or GARDIAN System. The GARDIAN System performs radiological assays of bulk shipping containers. The GARDIAN System is a cost effective and efficient means to ensure all non-radiological waste and recyclable materials arising from the decommissioning and demolition of the SONGS' site comply with all applicable regulatory requirements.

Hazardous and Industrial Waste Disposal

Uncontaminated lead shielding remaining after shutdown was assumed to be removed from its installed locations and shipped offsite by entities having a need for the material. The entities will receive the lead at no charge in return for providing the removal and shipping services. Non-Radioactive contaminated surfaces coated with tightly adhering and undamaged lead based paint will be removed as non-hazardous building demolition debris. All other chemicals and hazardous materials present at shutdown will be removed and properly disposed of during decommissioning.

3.5 Final Status Survey

The cost of performing a final status survey (FSS) is based on NUREG-1575, "Multi-Agency Radiation Survey and Site Investigation Manual (MARSSIM)," (Ref. No. 8). Estimates of MARSSIM Class I, II, and III survey designations are based on radiological assumptions regarding contamination resulting from small and large component removal activities. The FSS activity cost calculation includes the in-place remote survey of underground metal and concrete pipe, soil, and groundwater sampling and analysis. Estimated costs for NRC and Oak Ridge Institute for Science and Education (ORISE) verification are also included, and the NRC review period is incorporated into the project schedule.

3.6 Contingency

Contingencies are applied to cost estimates primarily to allow for unknown or unplanned occurrences during the actual program, e.g., increased radioactive waste materials volumes over that expected; equipment breakdowns, weather delays, and labor strikes. This is consistent with the definition provided in the DOE Cost Estimating Guide, DOE G 430.1-1, 3-28-97 (DOE G) (Ref. No. 9). Contingency "covers costs that may result from incomplete design, unforeseen and unpredictable conditions, or uncertainties within the defined project scope. The amount of contingency will depend on the status of design, procurement, and construction; and the complexity and uncertainties of the component parts of the project. Contingency is not to be used to avoid making an accurate assessment of expected costs." Energy*Solutions* determines site-specific contingency factors to be applied to each estimate based on industry practices.

The DOE has established a recommended range of contingencies as a function of completeness of program design, DOE G. The ranges are:

Type of Estimate	Contingency Range as a % of Total Estimate
Planning Phase Estimate	20-30
Budget Estimate	15-25
Title I (Preliminary Design Estimate)	10-20
Title II (Definitive Design Estimate)	5-15

Also, the Pacific Gas & Electric Company (PG&E) Technical Position Paper "Establishing an Appropriate Contingency Factor for Inclusion in the Decommissioning Revenue Requirements" (Ref. No. 13) was developed to review and determine a "conservative contingency factor" to be applied to decommissioning cost estimates. In that study it was determined that "based on an understanding of the level of project definition, and the extent and maturity of estimate input information used to develop decommissioning cost estimates, the 25 percent contingency factor is within the range of industry recognized cost engineering practices."

The contingencies presented in this study are consistent with the values presented in DOE G 430.1-1 for a Planning Phase estimate (Ref. No. 9) and the PG&E study (Ref. No. 13). As directed by SCE, Energy*Solutions* has applied a 25% contingency to all costs in this study, with the exception of following:

2013 and 2014 Actual Expenditures	0%
Department of Navy Easement Payments	15%
Hazardous and Asbestos Wastes	50%
Site Characterization Surveys	15%
Temporary Facilities	15%
Backfill and Compaction	15%

A reactor decommissioning program will be conducted under an NRC-approved Quality Assurance Program which meets the requirements of 10 CFR 50, Appendix B. However, the development of the quality assurance program, the performance of work under that program, and the effort required to ensure compliance with the program, is already included in the detailed cost estimate. Therefore, Energy*Solutions* does not include quality assurance as an element of the contingency allowance. The same is true for contamination. Where radioactive contamination or activated materials are dealt with, the Energy*Solutions* UCFs and associated calculations fully reflect the cost impact of that material, and a separate contingency is not required specifically due to working with contamination.

3.7 Cost Reporting

Total project costs are aggregated from the distributed activity and undistributed costs into the following categories – Labor, Materials and Equipment, Waste Disposal, and Other costs. Other costs include property taxes, insurance, license fees, permits, and energy. Waste Disposal costs are the summation of packaging, transportation, base disposal rate, and any applicable surcharges. Health physics (HP) supplies and small tool costs are calculated as a component of each distributed activity cost and included in the category of Material and Equipment, with the exception that HP supplies for the Utility HP staff are calculated and reported as an undistributed line item. A line item specific contingency is then calculated for each activity cost element.

4.0 SITE SPECIFIC TECHNICAL APPROACH

4.1 Facility Description

The San Onofre Nuclear Generating Station Units 2 & 3 site is located in southern California on the shore of the Pacific Ocean, about 62 miles Southeast of Los Angeles and approximately 51 miles Northwest of San Diego. The station is located entirely within the Camp Pendleton Marine Corps Base. The current Grant of Easement for the site from the United States Department of the Navy is currently scheduled to expire May 12, 2023 (Ref. No. 14). Units 2 & 3 occupy 52.8 acres of the 84 acre site. Approximately 16 acres are occupied by the North Industrial Area (formerly Unit 1), which is where the existing ISFSI is located.

The Nuclear Steam Supply System (NSSS) for both units are identical, with two independent loops, and utilizing pressurized light water cooled reactors (PWRs) supplied by Combustion Engineering, Inc. The construction permit was issued for an initial reactor power of 3,390 MWt with licensed Rated Thermal Power of 3,438 MWt.

The facility currently has an existing ISFSI containing spent fuel that was transferred into MPCs to maintain full core offload capability during operations and to facilitate decommissioning of Unit 1. This study also assumes that the MPCs will be licensed under a 10 CFR Part 72 general license, using the manufacturer's Certificate of Compliance. The 10 CFR Part 50 license will be maintained until decommissioning is complete and all spent fuel has been transferred to DOE.

Appendix A provides a list of the SONGS Unit 2 & 3 systems and structures included in the material inventory for this study.

4.2 Decommissioning Periods

The project periods consist of six License Termination periods, seven Spent Fuel Management periods (two of which are ISFSI decontamination and demolition periods), and six Site Restoration periods. As shown in Figure 1-1 above, the periods for each of these project areas are independent from (do not compete with) the periods for the other project areas. The project periods defined for this site-specific study and the major activities performed during each period are as follows:

License Termination Periods

Decon Pd 1 –Transition to Decommissioning

- Defuel Reactors
- Notification of Permanent Fuel Removal
- Disposition of LLRW Resins

Decon Pd 2 –Decommissioning Planning and Site Modifications

- Preparation of Decommissioning License Documents
- Preparation of NRC Deliverables
- Submit PSDAR to NRC
- Perform Historical Site Assessment and Site Characterization
- Planning, Design, and Implementation of Cold & Dark (Site Repowering)

- Design and Implement Spent Fuel Pool Support System Modifications, Control Room Relocation, and Spent Fuel Security System Modifications

- Select Decommissioning General Contractor (DGC)

Decon Pd 3 – Decommissioning Preparations and Reactor Internal Segmentation
- DGC Mobilization and Planning
- System Decontamination
- Reactor Internals Removal Preparations
- Reactor Internals Segmentation Planning and Implementation
- Purchase Dry Storage Modules for GTCC Waste
- Segment and Package Reactor Internals for Storage in the ISFSI

Decon Pd 4 – Plant Systems and Large Component Removal
- Upgrade Rail Spur on 'Owner Controlled Area' (does not affect spur connecting to CALTRANS).
- Install Large Array Radiation Detection System
- Remove, Package, and Dispose of Non-Essential Systems
- Asbestos and Lead Abatement
- Fuel Pool Closure
- Remove Spent Fuel Racks, Spent Fuel Pool Island Equipment, and Bridge Cranes
- Remove and Dispose of Legacy Class B & C Wastes
- Remove, Package, and Dispose of Essential Systems
- Removal and Disposal of Spent Resins, Filter Media, and Tank Sludge
- Large Component Removal
- Prepare License Termination Plan

Decon Pd 5 – Building Decontamination
- Decon Containment Buildings – Units 2 & 3
- Decon Turbine Buildings – Units 2 & 3
- Decon Fuel Handling Buildings – Units 2 & 3
- Decon Auxiliary Radwaste Building
- Decon Auxiliary Control Building
- Decon Penetration Buildings – Units 2 & 3
- Decon Safety Equipment and Main Steam Isolation Valve Buildings – Units 2 & 3
- Radiological Survey of Structures During Decon

Decon Pd 6 – License Termination During Decommissioning
- Final Status Survey
- ORISE Verification and NRC Approval

Spent Fuel Management Periods

SNF Pd 1 – Spent Fuel Transfer Management Transition
- Implementation of Security Enhancements Required for Reductions in Staff
- Cyber Security Modifications
- Post Fukushima Modifications – Unit 2
- Design and Fabricate Spent Fuel Canisters

SNF Pd 2 – Spent Fuel Transfer to Dry Storage
- Prepare Irradiated Fuel Management Plan
- Select Dry Storage System Canister Design and Vendor
- Design and Construct ISFSI Expansion
- Purchase, Deliver and Load Spent Fuel Canisters and Transfer to ISFSI

SNF Pd 3 – Dry Storage During Decommissioning Units 1, 2, & 3

SNF Pd 4 – Dry Storage Only – Units 1, 2, & 3

SNF Pd 5 – Dry Storage Only – Units 2, & 3

SNF D&D Pd 1 – ISFSI License Termination
- Preparation and NRC Review of License Termination Plan

SNF D&D Pd 2 – ISFSI Demolition
- Verification Survey of Horizontal Storage Modules
- Clean Demolition of ISFSI AHSMs and Pads
- Clean Demolition of ISFSI Support Structures
- Restore ISFSI Site
- Preparation of Final Report on Decommissioning and NRC Review

Site Restoration Periods

SR Pd 1 –Transition to Site Restoration
- Severance Costs from Post-Shutdown Reduction in Staffing
- Phase I and II Environmental Assessment of the Mesa Site
- Disposition of Hazardous Waste at the Mesa Site
- Site Characterization of the Mesa Site

SR Pd 2 –Building Demolition During Decommissioning
- Demolish South Access for Decommissioning, South Yard Facility, and Mesa Structures
- Finish Grade and Re-vegetate Mesa Site
- Mesa Lease Termination

SR Pd 3 – Subsurface Demolition Engineering & Permitting
- Hydrogeologic Investigation and Outfall Conduit Survey
- Subsurface Structure Removal Analyses for Lease Termination Activities
- Final Site Grading and Shoreline Protection Engineering Planning and Design
- Obtain Permits and Approvals

SR Pd 4 – Building Demolition to 3 Feet Below Grade
- Demolition Preparations
- De-Tension and Remove Containment Building Tendons – Units 2 & 3
- Demolish Diesel Generator Buildings – Units 2 & 3
- Demolish Condensate Buildings and Transformer Pads – Units 2 & 3

- Demolish Full Flow Areas and Turbine Buildings – Units 2 & 3
- Demolish Auxiliary Radwaste Building
- Demolish Auxiliary Control Building
- Remove Systems and Demolish Make-up Demineralizer Structures
- Demolish Penetration Buildings – Units 2 & 3
- Demolish Safety Equipment and Main Steam Isolation Valve Buildings – Units 2 & 3
- Demolish Fuel Handling Buildings to 3 Feet Below Grade – Units 2 & 3
- Demolish Containment Buildings to 3 Feet Below Grade – Units 2 & 3
- Demolish Intake and Discharge Structures to 3 Feet Below Grade

SR Pd 5 – Subgrade Structure Removal Below – 3 Feet

- Install Sheet Piling and Excavation Shoring, Dewatering System, and Effluent Treatment and Discharge Controls
- Demolish and Backfill Unit 3 Subsurface Structures
- Demolish and Backfill Unit 2 Subsurface Structures
- Demolish and Backfill Common Subsurface Structures
- Demolish and Backfill Intake Structure Inside Seawall Below -3 Feet
- Remove Off Shore Intake and Outfall Conduits
- Remove Sheet Piling, Excavation Shoring, and Dewatering and Effluent Treatment
- Finish Grading and Re-vegetate Site

SR Pd 6 – Final Site Restoration and Easement Termination

- Obtain Required Permits and Approvals
- Install Dewatering System and Effluent Treatment and Discharge Controls
- Remove and Stockpile Existing Seawall Erosion Protection
- Remove Unit 2 & 3 Seawall and Pedestrian Walkway
- Remove Remaining Intake Structure Beneath Seawall
- Backfill and Compaction of Excavation
- Remove Dewatering System & Effluent Treatment
- Remove Railroad Tracks, Gunite Slope Protection, Access Road, and North Parking Lot
- Finish Grading and Re-vegetate Site

4.3 Decommissioning Staff

Energy*Solutions* developed staffing based on the assumption that decommissioning will be performed by an experienced and qualified DGC, with oversight and management of the decommissioning operations performed by the Utility (Licensee) staff. It is also assumed that the Utility staff will be supplemented by a professional consulting engineering firm, particularly in the planning and preparation phase. The sizes of the Utility (Licensee) and DGC staffs are varied in each period in accordance with the requirements of the work activities. Details on the staff levels, by functional group, during each period are provided in Section 6.0.

4.4 Spent Fuel Management Staff

The largest spent fuel staff is in place while the fuel pool is operational during the spent fuel cooling period and the fuel assemblies are being transferred to dry storage. After all spent fuel

has been removed from the spent fuel pool, the staff is reduced. During spent fuel pool operations and the dry storage period, the full-time spent fuel management staff is supplemented with part-time staff to support fuel movements. Details on the staff levels, by functional group, during each period are provided in Section 6.0.

4.5 Spent Fuel Shipments

The spent fuel shipping schedules are based in part on the DOE's "Acceptance Priority Ranking & Annual Capacity Report," dated July 2004. (Ref. No. 12). The information regarding existing fuel inventory, planned transfers to dry storage and DOE's projected date of 2024 for acceptance of spent fuel is based on information provided by SCE. The spent fuel shipping schedule is provided in Appendix B. The spent fuel shipment schedule is based upon best current information and assumptions, as qualified and described elsewhere in this study, including in Section 2.2 above.

5.0 BASES OF ESTIMATE AND KEY ASSUMPTIONS

The bases of, and key assumptions for, this site-specific decommissioning estimate are presented below:

1. SCE's actual decommissioning expenses incurred from the time of permanent cessation of operations on June 7, 2013 until December 31, 2013 are included in the estimate. All other decommissioning cost data used in this study is current as of 2014. Totals and subtotals have been rounded to significant figures.

2. Energy*Solutions* developed a prompt dismantlement (DECON) project schedule based on a permanent shutdown date of June 7, 2013.

3. The decommissioning will be performed using currently available technologies.

4. DOE currently has no plans, program, or schedule in place for acceptance of utility spent fuel. However, for purposes of this decommissioning cost estimate, certain simplifying assumptions must be made regarding the schedule and rate of DOE performance. Therefore, while DOE's Standard Contract governing the acceptance of SCE's spent fuel allows for alternative removal schedules, including priority for shutdown reactors and exchanges of allocations, for purposes of this estimate DOE acceptance from the industry is assumed to commence in 2024 in accordance with SCE testimony to the Public Utilities Commission of the State of California (Ref. No. 17). The spent fuel shipment schedules are based upon the assumption that the DOE will accept spent fuel at the rate published in DOE's July 2004 Acceptance Priority Ranking & Annual Capacity Report (DOE/RW-0567) (Ref. No. 12). Additionally, SCE is reviewing available information from DOE to determine if the DOE start date assumption requires updating. The DCE will be revised accordingly as new information becomes available.

5. This estimate is based on site-specific building inventories and plant systems, as provided by Energy*Solutions*.

6. All transformers on site following shutdown are assumed to be polychlorinated biphenyl (PCB)-free, therefore, this study does not include costs for disposition of PCB contaminated transformers.

7. Cost for transportation of clean scrap metal to a recycler is included in the estimate; however, no credit is taken for the value of the scrap metal. Concrete debris and all other demolition debris is assumed to be removed from the site and disposed of at an out of state Class III landfill, consistent with the Governor of the State of California Executive Order D-62-02 (Ref. No. 16). The cost of installation and operation of Energy*Solutions*' GARDIAN system for bulk radiological assay of all wastes and recyclable materials leaving the SONGS site is included in the estimate. The purpose of the GARDIAN system is to ensure all materials not intended for disposal at a licensed facility meet all applicable requirements.

8. The estimate is based on final site restoration, in which all existing and proposed structures, with the exception of the switchyard, will be removed. Clean demolition costs are based on the assumption that all site improvements will be removed in their entirety. Clean backfill will be imported and placed to re-establish grade. The entire disturbed area of the site is to be graded, to restore the natural grade to the extent possible, and seeded.

9. Uncontaminated lead shielding remaining is assumed to be removed from its installed locations and shipped offsite by entities having a need for the material. The entities receive the lead at no charge in return for providing the removal and shipping services.

10. Site-specific information regarding contaminated soil was used as a basis for calculation of current costs for their remediation. While no known radiological or chemical remediation is required at the switchyard or the Mesa, those areas will be addressed as part of the Baseline Characterization Survey and Historical Site Assessment. If the studies conclude that radiological or chemical remediation is required at the switchyard or the Mesa, the DCE will be amended. For radiological contamination found at either the switchyard or the Mesa, the DCE will be amended to include all subsequent cost estimates for the remediation, which will be paid for by the SONGS participants in accordance with their cost allocations for the 'Common Facilities'. Chemical remediation of the switchyards will be paid by either SCE or SDG&E owners of the respective switchyards.

11. Costs for hazardous waste disposal, as well as asbestos and lead abatement, are included in this study.

12. All Class A waste is assumed to be disposed of at Energy*Solutions*' facility in Clive, Utah, in accordance with the existing Life-of-Plant Disposal Agreement between Energy*Solutions* and Southern California Edison, dated January 18, 2014 (Ref. No. 7). The following 2014 disposal rates will be applied:

    Demolition Debris and Soil - $57.97/Cubic Foot plus 5% Utah taxes
    Oversized Debris - $111.31/Cubic Foot plus 5% Utah taxes
    Containerized Waste Facility - $214.50/Cubic Foot plus 12% Utah taxes
    Large Components - $289.87/Cubic Foot plus 5% Utah taxes
    Cask Shipments - $44,059/Cask plus 12% Utah taxes

    Class A waste includes Dry Active Waste (DAW) arising from the disposal of contaminated protective clothing and health physics supplies.

13. Class B, C, and GTCC waste disposal costs are based on recent quotes for disposal of activated hardware and resins at the WCS facility. All resins and filter waste is assumed to be Class B.

14. Shipping costs for the Class B and C waste are based on a distance of 1,079 miles one way from SONGS to the WCS site.

15. GTCC is not subject to the same storage and security requirements as spent fuel and therefore is not required to be stored on the ISFSI pad. But for purposes of this estimate and to facilitate decommissioning, GTCC waste generated from the segmentation of the reactor internals is assumed to be packaged in Dry Shielded Canisters (DSCs) and placed in Advanced Horizontal Storage Modules (AHSMs) in the ISFSI to await final disposition at a DOE repository.

16. It is assumed that a total of six DSCs per unit will be required for GTCC waste.

17. Reactor vessel and internals curie estimates were derived from the values for the Reference PWR vessel and internals in NUREG/CR-0130 (Ref. No. 5). These values were adjusted for decay period.

18. The Energy*Solutions* site-specific classification of radioactive wastes for the SONGS Plant identified that the spent fuel assemblies and two components within the reactor vessel (the Core Shroud Assembly and the Lower Core Grid Plate) will exceed Class C limitations.

19. The spent fuel shipments are based upon best current information and assumptions, as qualified and described elsewhere in this study, including in Section 2.2. above.

20. Spent fuel will remain in the spent fuel pool for six years before being transferred to the ISFSI.

21. The costs for ISFSI construction and transfer of spent fuel from Units 2 & 3 to dry storage were developed by SCE and furnished to Energy*Solutions*. Following completion of spent fuel transfers to dry storage the cost of maintenance and operation of the ISFSI is distributed between Units 1, 2 and 3 based on the relative percentages of spent fuel assemblies in storage. The percentages are 10, 45, and 45 for Units 1, 2, and 3, respectively. The exception is that all property taxes are solely the liability of Units 2 & 3. Following completion of SNF Pd 4 – Dry Storage Only Units 1, 2, and 3, all ISFSI maintenance and operating costs are assigned to Units 2 & 3 until the ISFSI D&D. During ISFSI D&D costs are distributed to all three units in the same percentages of 10, 45, and 45.

22. DOE has not committed to accept SCE's canistered spent fuel. But for purposes of this estimate, it is assumed that an SCE-funded dry storage facility will not be necessary.

23. Costs for ISFSI demolition are included in this estimate. SCE assumes that portions of the AHSM concrete will be activated.

24. Energy*Solutions* has assumed that the 10 CFR Part 50 license will be maintained until DOE has taken possession of the spent fuel.

25. SCE's annual ISFSI insurance premiums of $302,000 are assumed to be incurred until all fuel shipments have been completed and the structure is no longer in use.

26. SCE's Emergency Preparedness (FEMA) fees of $500,000 per year and California Office of Emergency Services fees of $2,800,000 per year are applied until the spent fuel pool is empty. These fees were supplied by SCE.

27. SCE's current annual property taxes are assumed to be reduced to a constant $1,500,000 per year. The property taxes are a license termination expense until the completion of decommissioning, and then a spent fuel management expense until completion of the ISFSI D&D.

28. Energy*Solutions* has included the annual NRC 10 CFR 171.15(c)(2) fees, for reactors in decommissioning of $231,000/yr per unit until decommissioning is completed as a license termination expense. Following completion of decommissioning, this expense is continued as a spent fuel management cost for maintenance of the 10 CFR Part 50 license.

29. Energy*Solutions* has included Environmental Permits and Fees of $1,900,000 per year as supplied by SCE.

30. Energy*Solutions* has included NRC inspection fees during each decommissioning period based on the type and level of activities being performed.

31. SONGS annual insurance premiums, in 2014 dollars as supplied by SCE, are as follows:

    Nuclear Property Primary - $4,878,099
    Nuclear Liability - $1,151,075
    Additional Liability, Non-Nuclear - $3,576,519
    Workers' Compensation - $180,335
    Property Insurance - $353,286

    The premium amounts have been adjusted by Energy*Solutions* in accordance with information furnished by SCE to meet the requirements of each period.

32. Site operating expenses expected to be incurred during decommissioning and spent fuel management are included in the estimate. These costs include materials and services, utilities (water, gas, phone), telecommunications equipment, non-process computers, personal computers and tools and equipment. These costs were calculated based on information provided by SCE and adjusted by Energy*Solutions* to match the requirements of each period, based on staffing levels.

33. Site Lease and Easement expenses of $2,300,000 per year until the Mesa lease is terminated are included in the estimate. Following termination of the Mesa lease the site lease and easement expenses are reduced to $299,920 per year. These costs are based on information provided by SCE.

34. Utility (Licensee) staff positions and average direct burdened salary (i.e. total compensation) data in 2014 dollars were supplied by SCE.

35. Severance costs for those employees terminated as a result of SONGS decommissioning, including those costs required under California law are included in the estimate. Severance costs for Reductions-in-Force (RIFs) that occurred immediately after shutdown, and during the course of spent fuel management and decommissioning are assumed to be a site restoration expense and are included in the estimate.

36. Severance costs per employee were provided by SCE.

37. DGC staff salaries, including overhead and profit, were determined by Energy*Solutions* and represent Energy*Solutions*' standard assumptions for these rates.

38. The professional personnel used for the planning and preparation activities, and DGC personnel, are assumed to be paid per diem at the rate of $204/day, based on per diem rates from U.S. General Services Administration (GSA) for Orange County, California.

39. Craft labor rates were taken from the CA Union Craft Rate Sheet, dated January 9, 2014. Craft labor rates for disciplines not provided in the rate sheet have been taken from the 2014 RS Means Labor Rates for the Construction Industry (Ref. No. 10), for Anaheim, CA. Since the skilled laborers are assumed to be supplied by the local union hall, they will not be paid per diem.

40. The security guard force included in this estimate has been sized in accordance with the current Design Basis Threat assessment.

41. This study follows the occupational exposure principles of As Low As Reasonably Achievable (ALARA) through the use of productivity loss factors that incorporate such items as the use of respiratory protection and personnel protective clothing. These factors increase the work duration and cost.

42. The costs of all required safety analyses and safety measures for the protection of the general public, the environment, and decommissioning workers are included in the cost estimates. This reflects the requirements of:

10 CFR 20	Standards for Protection Against Radiation
10 CFR 50	Domestic Licensing of Production and Utilization Facilities
10 CFR 61	Licensing Requirements for Land Disposal of Radioactive Waste
10 CFR 71	Packaging of Radioactive Material for Transport
10 CFR 72	Licensing Requirements for the Independent Storage of Spent Nuclear Fuel and High-Level Radioactive Waste
29 CFR 1910	Occupational Safety and Health Standards

49 CFR 170-189	Department of Transportation Regulations Governing the Transport of Hazardous Materials
Reg. Guide 1.159	Assuring the Availability of Funds for Decommissioning Nuclear Reactors

43. Activity labor costs do not include any allowance for delays between activities, nor is there any cost allowance for craft labor retained on site while waiting for work to become available.

6.0 STUDY RESULTS

This study analyzes the following technical approach to decommissioning as defined by SCE:

- Prompt DECON methodology.
- Permanent cessation of operations and commencement of decommissioning planning on June 7, 2013.
- Termination of spent fuel pool operation six years after permanent shutdown.
- Spent fuel will be stored in MPCs at an on-site ISFSI.
- A dry transfer facility will not be necessary for transfer of SNF for transport.
- Decommissioning will be performed by a DGC with oversight by the SONGS participants.
- LOP Disposal Rates are used for Class A LLRW.
- WCS Texas Disposal Rates are used for Class B and C LLRW.
- DOE begins accepting spent fuel from the industry in 2024.

Spent Fuel Shipping Schedule

The spent fuel shipping schedule is provided in Appendix B. Spent fuel shipments from the industry to DOE will begin in 2024. The spent fuel shipment schedules are based upon best current information and assumptions, as qualified and described elsewhere in this study, including in Section 2.2 above.

Cost and Schedule

Figure 6-1 is a summary project schedule. A detailed schedule is provided in Appendix C. Table 6-1 summarizes the period durations and total costs, including contingency, for License Termination, Spent Fuel, and Site Restoration activities. A detailed cost table is provided in Appendix D, and a table of annual expenditures is provided in Appendix E.

Project Staffing

This scenario is based on the assumption that decommissioning will be performed by an experienced and qualified DGC, with oversight and management of the decommissioning operations performed by the Licensee staff. Utility (Licensee) staffing levels, by organizational department and function, for each period are provided in Table 6-2. The DGC staffing levels, by organizational department and function, for each period are provided in Table 6-3.

LLRW Disposal Volumes

LLRW disposal is a significant element of the decommissioning project. The estimated cubic feet of waste are summarized as follows:

Waste Class	Unit 2	Unit 3	Total
Class A	1,832,961	1,819,680	3,652,641
Class B	7,600	7,600	15,200
Class C	4,095	4,095	8,190
GTCC	941	941	1,882

Waste disposal volumes and costs, itemized by packaging, transportation, surcharges and disposal costs by waste class and facility, are provided in Table 6-4. The waste disposal costs provided in Table 6-4 do not include contingency.

Figure 6-1
Summary Schedule

DECON with Dry Storage, 2013 Shutdown and DOE Acceptance in 2024

Task Name	Start	Finish
Post-Shutdown Spent Fuel Management	06/07/2013	09/08/2051
Spent Fuel Shipping Complete - Unit 1	12/31/2035	12/31/2035
Spent Fuel Shipping Complete - Units 2 & 3	12/31/2049	12/31/2049
SNF Pd 1 - Spent Fuel Management Transition	06/07/2013	12/31/2013
SNF Pd 2 - Spent Fuel Transfer to Dry Storage	1/1/2014	06/01/2019
SNF Pd 3 - Dry Storage During Decommissioning - Units 1, 2 and 3	06/01/2019	12/05/2031
SNF Pd 4 - Dry Storage Only - Units 1, 2 and 3	12/05/2031	12/31/2035
SNF Pd 5 - Dry Storage Only - Units 2 and 3	12/31/2035	12/31/2049
SNF D&D Pd 1 - ISFSI D&D Planning	12/31/2049	05/06/2050
SNF D&D Pd 2 - ISFSI D&D	05/06/2050	09/08/2051
Part 50 License Termination	06/07/2013	12/24/2032
Announcement of Cessation of Operations (June 7, 2013)	06/07/2013	06/07/2013
Decon Pd 1 - Transition to Decommissioning	06/07/2013	12/31/2013
Decon Pd 2 - Decommissioning Planning and Site Modifications	1/1/2014	06/30/2015
Decon Pd 3 - Decommissioning Preparations and Reactor Internals Segmentation	06/30/2015	06/01/2019
Decon Pd 4 - Plant Systems and Large Component Removal	06/01/2019	09/24/2022
Decon Pd 5 - Building Decontamination	09/24/2022	07/13/2024
Decon Pd 6 - License Termination During Demolition	07/13/2024	12/24/2032
Site Restoration	06/07/2013	12/15/2051
SR Pd 1 - Transition to Site Restoration	06/07/2013	06/30/2015
SR Pd 2 - Building Demolition During Decommissioning	06/30/2015	07/11/2017
SR Pd 3 - Subsurface Demolition Engineering and Permitting	10/01/2019	07/13/2024
SR Pd 4 - Building Demolition to 3 Feet Below Grade	07/13/2024	10/14/2028
SR Pd 5 - Subgrade Structure Removal Below -3 Feet	10/14/2028	12/05/2031
SR Pd 6 - Final Site Restoration and Lease Termination	05/06/2050	12/15/2051
Final Easement Termination	12/15/2051	12/15/2051

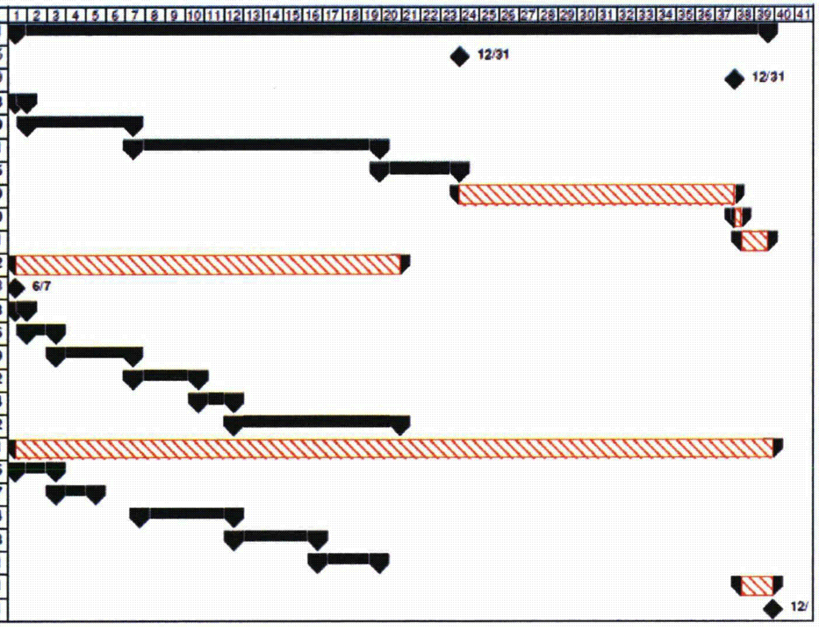


Table 6-1[3]
Cost and Schedule Summary
(2014 Dollars in Thousands)

Period No.	Period Description	Start	End	Years	Unit 2 Cost	Unit 3 Cost	Total Cost
License Termination (50.75(c))							
Decon Pd 1	Transition to Decommissioning	6/7/2013	12/31/2013	0.56	$25,749	$26,566	$52,315
Decon Pd 2	Decommissioning Planning and Site Modifications	1/1/2014	6/30/2015	1.49	$118,709	$122,430	$241,140
Decon Pd 3	Decommissioning Preparations and Reactor Internals Segmentation	6/30/2015	6/1/2019	3.92	$262,210	$276,799	$539,009
Decon Pd 4	Plant Systems and Large Component Removal	6/1/2019	9/24/2022	3.31	$392,029	$412,475	$804,504
Decon Pd 5	Building Decontamination	9/24/2022	7/13/2024	1.80	$212,447	$216,659	$429,106
Decon Pd 6	License Termination During Demolition	7/13/2024	12/24/2032	8.44	$23,085	$23,085	$46,171
Account Total				**19.52**	**$1,034,230**	**$1,078,016**	**$2,112,246**
Spent Fuel (50.54(bb)) and (72.30)							
SNF Pd 1	Spent Fuel Management Transition	6/7/2013	12/31/2013	0.56	$63,891	$66,105	$129,997
SNF Pd 2	Spent Fuel Transfer to Dry Storage	1/1/2014	6/1/2019	5.41	$344,629	$372,193	$716,822
SNF Pd 3	Dry Storage During Decommissioning - Units 1, 2 and 3	6/1/2019	12/5/2031	12.51	$61,425	$61,425	$122,849
SNF Pd 4	Dry Storage Only - Units 1, 2 and 3	12/5/2031	12/31/2035	4.07	$29,383	$29,383	$58,765
SNF Pd 5	Dry Storage Only - Units 2 and 3	12/31/2035	12/31/2049	14.00	$107,326	$107,326	$214,653
SNF D&D Pd 1	ISFSI License Termination	12/31/2049	5/6/2050	0.34	$1,260	$1,260	$2,520
SNF D&D Pd 2	ISFSI Demolition	5/6/2050	9/8/2051	1.34	$15,295	$15,295	$30,590
Account Total				**38.23**	**$623,209**	**$652,987**	**$1,276,196**
Site Restoration							
SR Pd 1	Transition to Site Restoration	6/7/2013	6/30/2015	2.06	$64,280	$66,210	$130,489
SR Pd 2	Building Demolition During Decommissioning	6/30/2015	7/11/2017	2.03	$13,003	$37,242	$50,245
SR Pd 3	Subsurface Demolition Engineering and Permitting	10/1/2019	7/13/2024	4.78	$15,593	$22,319	$37,912
SR Pd 4	Building Demolition to 3 Feet Below Grade	7/13/2024	10/14/2028	4.25	$124,953	$134,113	$259,066
SR Pd 5	Subgrade Structure Removal Below - 3 Feet	10/14/2028	12/5/2031	3.14	$171,987	$269,560	$441,547
SR Pd 6	Final Site Restoration and Lease Termination	5/6/2050	12/15/2051	1.60	$33,482	$70,064	$103,545
Account Total				**17.86**	**$423,297**	**$599,507**	**$1,022,804**
Grand Total					**$2,080,735**	**$2,330,511**	**$4,411,246**

[3] Rows and columns may not add correctly due to rounding.

Table 6-2
Utility Staff Levels

License Termination – 50.75(c) Utility Staff

Department	Decon Pd 1	Decon Pd 2	Decon Pd 3	Decon Pd 4	Decon Pd 5	Decon Pd 6
Decommissioning	0	21	21	25	18	0
Engineering	0	49	14	14	12	0
Maintenance and Work Control	0	38	10	10	3	0
Operations	0	15	7	7	0	0
Oversight and Nuclear Safety	0	7	2	2	1	0
Radiation Protection and Chemistry	0	27	26	31	26	0
Regulatory and Emergency Planning	0	10	4	4	4	0.5
Safety and Human Performance	0	13	7	7	7	0
Security Admin	0	6	6	6	6	0
Security Guard Force	0	12	12	12	12	0
Site Management and Administration	0	13	13	13	9	1
Period Totals	**0**	**211**	**122**	**131**	**98**	**1.5**

Spent Fuel - 50.54(bb) Utility Staff

Department	SNF Pd 1	SNF Pd 2	SNF Pd 3	SNF Pd 4	SNF Pd 5	SNF D&D Pd 1	SNF D&D Pd 2
Spent Fuel Shipping	0	0	0	2	2	0	0
Decommissioning	0	0	0	0	0	1	1
Engineering	0	1	1	1	1	0	1
Maintenance and Work Control	0	31	0	0	0	0	0
Operations	0	45	1	1	1	0	0
Oversight and Nuclear Safety	0	1	0.25	0.25	0.25	0	0
Radiation Protection and Chemistry	0	6	4	4	4	1	2
Regulatory and Emergency Planning	0	0	0	0	0	1	1
Security Admin	0	14	10	8	8	1	1
Security Guard Force	0	178	35	35	35	5	5
Site Management and Administration	0	0	0	0	0	1	1
Period Total	**0**	**276**	**51.25**	**54.25**	**54.25**	**10**	**12**

Site Restoration - Utility Staff

Department	SR Pd 1	SR Pd 2	SR Pd 3	SR Pd 4	SR Pd 5	SR Pd 6
Decommissioning	0	2	0	5	4	2
Engineering	0	1	0	2	1	0
Maintenance and Work Control	0	1	0	2	2	2
Regulatory and Emergency Planning	0	1	0	0	0	0
Safety and Human Performance	0	1	0	2	1	1
Security Admin	0	0	0	1	1	0
Security Guard Force	0	0	0	5	5	0
Site Management and Administration	0	0	0	4	3	3
Period Totals	**0**	**6**	**0**	**21**	**17**	**8**

Table 6-3
DGC Staff Levels

License Termination – 50.75(c) DGC Staff

Department	Decon Pd 3	Decon Pd 4	Decon Pd 5	Decon Pd 6
Administration	9	17	17	0
Engineering	15	29	14	0
Health Physics	16	73	73	2
Management	3	3	3	0
Quality Assurance	2	5	4	0
Waste Operations	7	16	16	0
Period Totals	**52**	**143**	**127**	**2**

Spent Fuel - 50.54(bb) - DGC Staff

Department	SNF D&D Pd 2
Administration	1
Engineering	2
Health Physics	3
Management	1
Quality Assurance	1
Waste Operations	4
Period Totals	**12**

Site Restoration DGC Staff

Department	SR Pd 1	SR Pd 2	SR Pd 3	SR Pd 4	SR Pd 5	SR Pd 6
Administration	0	0	0	10	5	4
Engineering	0	0	0	13	11	5
Health Physics	0	0	0	3	0	0
Management	0	0	0	2	2	1
Quality Assurance	0	0	0	2	1	0
Waste Operations	0	0	0	11	7	7
Period Totals	**0**	**0**	**0**	**41**	**26**	**17**

Table 6-4
Waste Disposal Volumes
(Cost Excludes Contingency - 2014 Dollars)

Facility and Waste Class	Waste Weight (LBs)	Waste Volume (CF)	Burial Volume (CF)	Packaging Cost	Transportation Cost	Base Burial Cost	Total Disposal Cost
Class B and C Facility							
Class B	1,132,323	6,696	15,199	$1,199,186	$6,433,599	$72,635,570	$80,268,355
Class C	407,380	1,546	8,191	$2,064,309	$26,706,007	$39,142,870	$67,913,186
GTCC	92,861	190	1,882	$196,288	$1,680,000	$38,775,980	$40,652,268
	1,632,564	**8,431**	**25,272**	**$3,459,782**	**$34,819,606**	**$150,554,420**	**$188,833,808**
Energy*Solutions*							
Class A – Debris	200,560,122	3,229,506	3,308,050	$3,804,262	$13,779,286	$211,423,909	$229,007,458
Class A – Oversize	9,967,521	146,943	184,730	$187,314	$784,285	$22,669,947	$23,641,545
Class A – Containerized Waste	1,053,914	12,287	16,303	$397,152	$364,322	$4,112,378	$4,873,851
Class A – Large Component	11,480,200	108,866	136,373	$6,313,568	$69,622,664	$43,582,464	$119,518,696
Class A – Mixed Waste	62,643	3,012	3,012	$67,887	$12,448	$801,226	$881,561
	223,124,400	**3,500,614**	**3,648,469**	**$10,770,182**	**$84,563,005**	**$282,589,924**	**$377,923,111**
Other							
Out of State Class III Landfill	1,909,207,440	25,212,269	29,372,422	$0	$146,326,469	$43,929,750	$190,256,219
Scrap Metal Recycler	184,787,372	377,117	7,391,495	$0	$911,926	$0	$911,926
	2,093,994,812	**25,589,386**	**36,763,917**	**$0**	**$147,238,394**	**$43,929,750**	**$191,168,144**
Grand Total	**2,318,751,776**	**29,098,431**	**40,437,658**	**$14,229,964**	**$266,621,006**	**$477,074,094**	**$757,925,064**

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Appendix A

List of Systems and Structures

SONGS Plant System and Structure List

Common

Type	System Name or Description
Non	Not Used
Struct	Administration Building (K-40/50)
Struct	AWS Building
Struct	Building L-50
Struct	Gunite Slope Protection
Struct	High Flow Make-Up Demineralizer Area
Struct	ISFSI Support Structures
Struct	Maintenance Building 1 (B-43/B-44)
Struct	Maintenance Building 2 (B-49/B-50)
Struct	Maintenance Building 4 (B-64/B-65)
Struct	Maintenance Building 5 (B-62/B-63)
Struct	Mesa Buildings
Struct	Not Used
Struct	Outage Control Center Building
Struct	REMS Staging Pad
Struct	Seawall - Units 2 & 3
Struct	Security Access Building (A-80, 81, 82)
Struct	Service Building (K-10, 20, 30)
Struct	South Security Processing Facility (K-70)
Struct	South Yard Facility Buildings (T-10, 20, 60 and Haz Mat.)
Struct	Staging Warehouse Building
Ess	Auxilary Control Systems - Unit 2
Ess	Fuel Handling Building Systems - Unit 2
Ess	Radwaste Systems - Unit 2
Non	Condenstate Storage Systems - Unit 2
Non	Containment Building Systems - Unit 2
Non	Diesel Generator Systems - Unit 2
Non	Full Flow Areas Systems - Unit 2
Non	Intake Systems - Unit 2
Non	Penetration Building Systems - Unit 2
Non	Safety Equipment Building Systems - Unit 2
Non	Turbine Bldg Equip to 9 ft - Unit 2
Struct	Condensate Storage Area - Unit 2
Struct	Containment Building - Unit 2
Struct	Control Building - Unit 2
Struct	Diesel Generator Building - Unit 2
Struct	Fuel Handling Building - Unit 2
Struct	Full Flow Building - Unit 2
Struct	Intake Structure - Unit 2
Struct	Penetration Building - Unit 2
Struct	Radwaste Building - Unit 2
Struct	Safety Equipment Building - Unit 2
Struct	Tunnels - Unit 2
Struct	Turbine Building - Unit 2
Ess	Auxilary Control Systems - Unit 3
Ess	Fuel Handling Building Systems - Unit 3

SONGS Plant System and Structure List

Unit 3

Type	System Name or Description
Ess	Radwaste Systems - Unit 3
Non	Condenstate Storage Systems - Unit 3
Non	Containment Building Systems - Unit 3
Non	Diesel Generator Systems - Unit 3
Non	Full Flow Areas Systems - Unit 3
Non	Intake Systems - Unit 3
Non	Penetration Building Systems - Unit 3
Non	Safety Equipment Building Systems - Unit 3
Non	Turbine Bldg Equip to 9 ft - Unit 3
Non	Turbine Generator to 63 ft - Unit 3
Struct	Condensate Storage Tank Area - Unit 3
Struct	Containment Building - Unit 3
Struct	Control Building - Unit 3
Struct	Diesel Generator Building - Unit 3
Struct	Fuel Handling Building - Unit 3
Struct	Full Flow Building - Unit 3
Struct	Intake Structure - Unit 3
Struct	Penetration Building - Unit 3
Struct	Radwaste Building - Unit 3
Struct	Safety Equipment Building - Unit 3
Struct	Tunnels - Unit 3
Struct	Turbine Building - Unit 3

Appendix B

Spent Fuel Shipping Schedule

SONGS Unit 2 & Unit 3
Spent Fuel Shipping Schedule
2024 DOE Acceptance

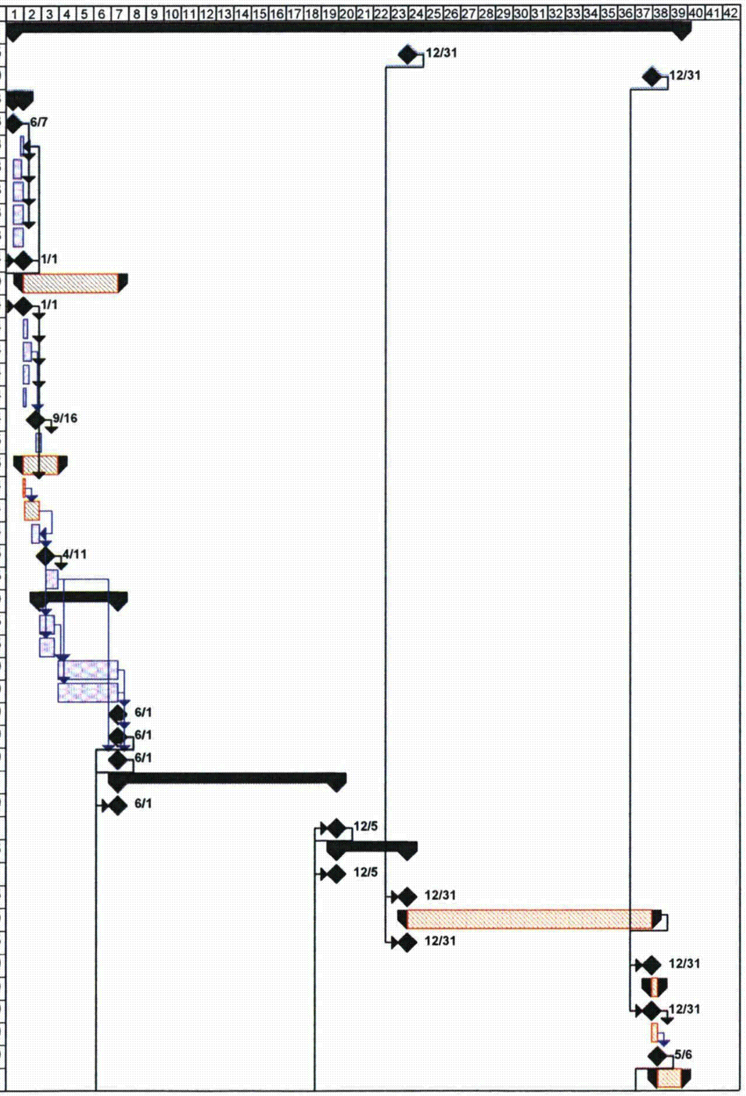
	On-Site Inventory (Beginning of Year)				On-Site Transfers (During Year)		Off-Site Transfers (During Year)			
Year	Units 2 & 3 Fuel Assemblies in Wet Storage	Units 2 & 3 Fuel Assemblies in Dry Storage	Units 2 & 3 Fuel Assemblies in On-Site Storage	Units 2 & 3 Canisters in ISFSI	Unit 2 & 3 Fuel Assemblies Transferred to ISFSI	Unit 2 & 3 Fuel Canisters Transferred to ISFSI	Unit 2 Assemblies Transferred to DOE	Unit 3 Assemblies Transferred to DOE	Units 2 & 3 Assemblies Transferred to DOE	Units 2 & 3 Canisters Transferred to DOE
2014	2668	792	3460	33	0	0	0	0	0	0
2015	2668	792	3460	33	0	0	0	0	0	0
2016	2668	792	3460	33	0	0	0	0	0	0
2017	2668	792	3460	33	768	24	0	0	0	0
2018	1900	1560	3460	57	1,536	48	0	0	0	0
2019	364	3096	3460	105	364	13	0	0	0	0
2020	0	3460	3460	118	0	0	0	0	0	0
2021	0	3460	3460	118	0	0	0	0	0	0
2022	0	3460	3460	118	0	0	0	0	0	0
2023	0	3460	3460	118	0	0	0	0	0	0
2024	0	3460	3460	118	0	0	0	0	0	0
2025	0	3460	3460	118	0	0	0	0	0	0
2026	0	3460	3460	118	0	0	0	0	0	0
2027	0	3460	3460	118	0	0	0	0	0	0
2028	0	3460	3460	118	0	0	0	0	0	0
2029	0	3460	3460	118	0	0	0	0	0	0
2030	0	3460	3460	118	0	0	48	48	96	4
2031	0	3364	3364	114	0	0	192	96	288	12
2032	0	3076	3076	102	0	0	120	120	240	10
2033	0	2836	2836	92	0	0	0	96	96	4
2034	0	2740	2740	88	0	0	112	120	232	8
2035	0	2508	2508	80	0	0	96	96	192	6
2036	0	2316	2316	74	0	0	128	96	224	7
2037	0	2092	2092	67	0	0	0	0	0	0
2038	0	2092	2092	67	0	0	96	128	224	7
2039	0	1868	1868	60	0	0	96	96	192	6
2040	0	1676	1676	54	0	0	96	96	192	6
2041	0	1484	1484	48	0	0	0	0	0	0
2042	0	1484	1484	48	0	0	96	96	192	6
2043	0	1292	1292	42	0	0	96	96	192	6
2044	0	1100	1100	36	0	0	96	96	192	6
2045	0	908	908	30	0	0	128	96	224	7
2046	0	684	684	23	0	0	96	128	224	7
2047	0	460	460	16	0	0	96	230	326	11
2048	0	134	134	5	0	0	0	0	0	0
2049	0	134	134	5	0	0	134	0	134	5
2050	0	0	0	0	0	0	0	0	0	0

Appendix C

Detailed Project Schedule

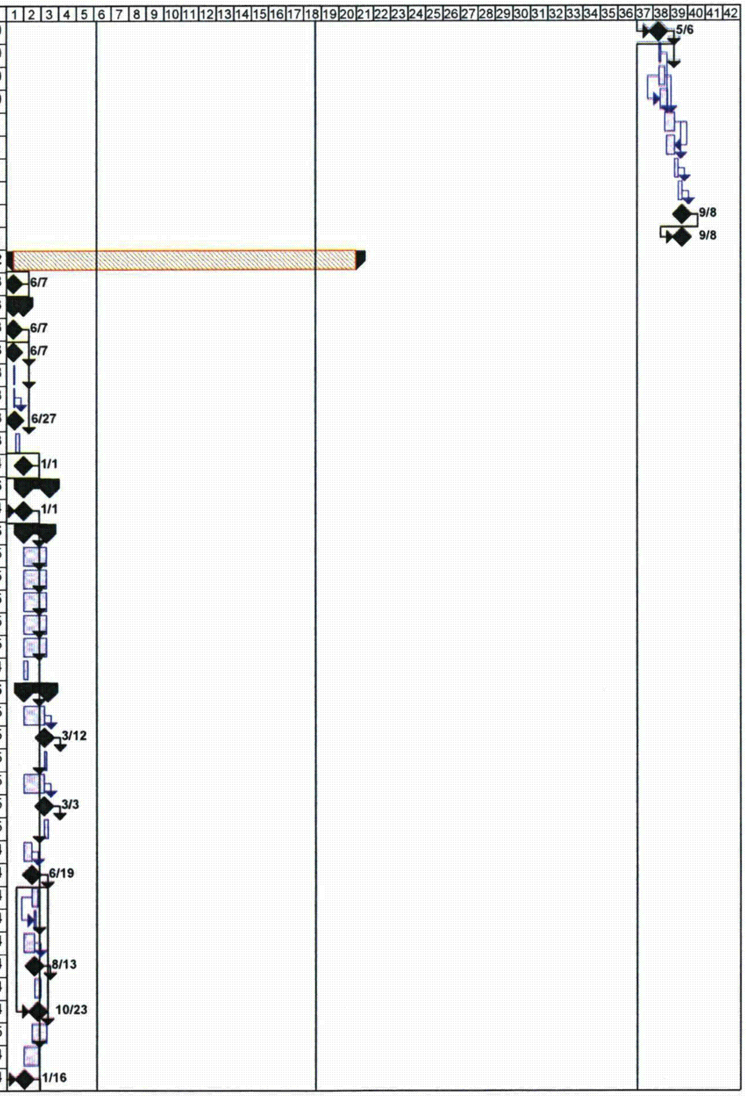
SONGS 2 & 3
Detailed Project Schedule
Prompt DECON, DOE Repository Opens 2024

ID	Task Name	Duration	Start	Finish
1	**Post-Shutdown Spent Fuel Management**	**9982 days**	**06/07/2013**	**09/08/2051**
2	Spent Fuel Shipping Complete - Unit 1	0 days	12/31/2035	12/31/2035
3	Spent Fuel Shipping Complete - Units 2 & 3	0 days	12/31/2049	12/31/2049
4	**SNF Pd 1 - Spent Fuel Management Transition**	**149 days**	**06/07/2013**	**12/31/2013**
5	SNF Pd 1 Begins	0 days	06/07/2013	06/07/2013
6	Security Enhancements	45 days	10/31/2013	12/31/2013
7	Design and Fabricate Spent Fuel Canisters	120 days	06/07/2013	11/21/2013
8	Post Fukushima Modifications - Unit 2	148 days	06/07/2013	12/31/2013
9	Post Fukushima Modifications - Unit 3	148 days	06/07/2013	12/31/2013
10	Cyber Security Modifications	148 days	06/07/2013	12/31/2013
11	SNF Pd 1 Ends	0 days	01/01/2014	01/01/2014
12	**SNF Pd 2 - Spent Fuel Transfer to Dry Storage**	**1413 days**	**1/1/2014**	**06/01/2019**
13	SNF Pd 2 Begins	0 days	01/01/2014	01/01/2014
14	Security Enhancements	60 days	01/02/2014	03/26/2014
15	Prepare Irradiated Fuel Management Plan	118 days	01/02/2014	06/14/2014
16	Decay Heat Analysis	85 days	01/02/2014	04/30/2014
17	Zirconium Fire and Shine Analysis	42 days	01/02/2014	02/28/2014
18	Issue Irradiated Fuel Management Plan	0 days	09/16/2014	09/16/2014
19	NRC Review of Irradiated Fuel Management Plan	82 days	09/17/2014	01/08/2015
20	**ISFSI Expansion**	**513 days**	**01/02/2014**	**12/19/2015**
21	Select Dry Storage System Canister Design and Vendor	30 days	01/02/2014	02/12/2014
22	ISFSI Pad Study	220 days	02/03/2014	12/06/2014
23	Design ISFSI Expansion	120 days	06/24/2014	12/06/2014
24	ISFSI Permit Approved By California Coastal Commission	0 days	04/11/2015	04/11/2015
25	Construct ISFSI Expansion	180 days	04/14/2015	12/19/2015
26	**Canister Purchase, Fabrication and Fuel Tranfer**	**1170 days**	**12/09/2014**	**06/01/2019**
27	Purchase Spent Fuel Canisters and AHSMs - Unit 2	220 days	12/09/2014	10/10/2015
28	Purchase Spent Fuel Canisters and AHSMs - Unit 3	220 days	12/09/2014	10/10/2015
29	Load Spent Fuel Canisters and Transfer to ISFSI - Unit 2	900 days	12/22/2015	06/01/2019
30	Load Spent Fuel Canisters and Transfer to ISFSI - Unit 3	900 days	12/22/2015	06/01/2019
31	Unit 2 Spent Fuel Pool Empty	0 days	06/01/2019	06/01/2019
32	Unit 3 Spent Fuel Pool Empty	0 days	06/01/2019	06/01/2019
33	SNF Pd 2 Ends	0 days	06/01/2019	06/01/2019
34	**SNF Pd 3 - Dry Storage During Decommissioning - Units 1, 2 and 3**	**3265 days**	**06/01/2019**	**12/05/2031**
35	SNF Pd 3 Begins	0 days	06/01/2019	06/01/2019
36	SNF Pd 3 Ends	0 days	12/05/2031	12/05/2031
37	**SNF Pd 4 - Dry Storage Only - Units 1, 2 and 3**	**1061 days**	**12/05/2031**	**12/31/2035**
38	SNF Pd 4 Begins	0 days	12/05/2031	12/05/2031
39	SNF Pd 4 Ends	0 days	12/31/2035	12/31/2035
40	**SNF Pd 5 - Dry Storage Only - Units 2 and 3**	**3654 days**	**12/31/2035**	**12/31/2049**
41	SNF Pd 5 Begins	0 days	12/31/2035	12/31/2035
42	SNF Pd 5 Ends	0 days	12/31/2049	12/31/2049
43	**SNF D&D Pd 1 - ISFSI D&D Planning**	**90 days**	**12/31/2049**	**05/06/2050**
44	SNF D&D Pd 1 Begins	0 days	12/31/2049	12/31/2049
45	Preparation and NRC Review of License Termination Plan	90 days	01/01/2050	05/06/2050
46	SNF D&D Pd 1 Ends	0 days	05/06/2050	05/06/2050
47	**SNF D&D Pd 2 - ISFSI D&D**	**350 days**	**05/06/2050**	**09/08/2051**



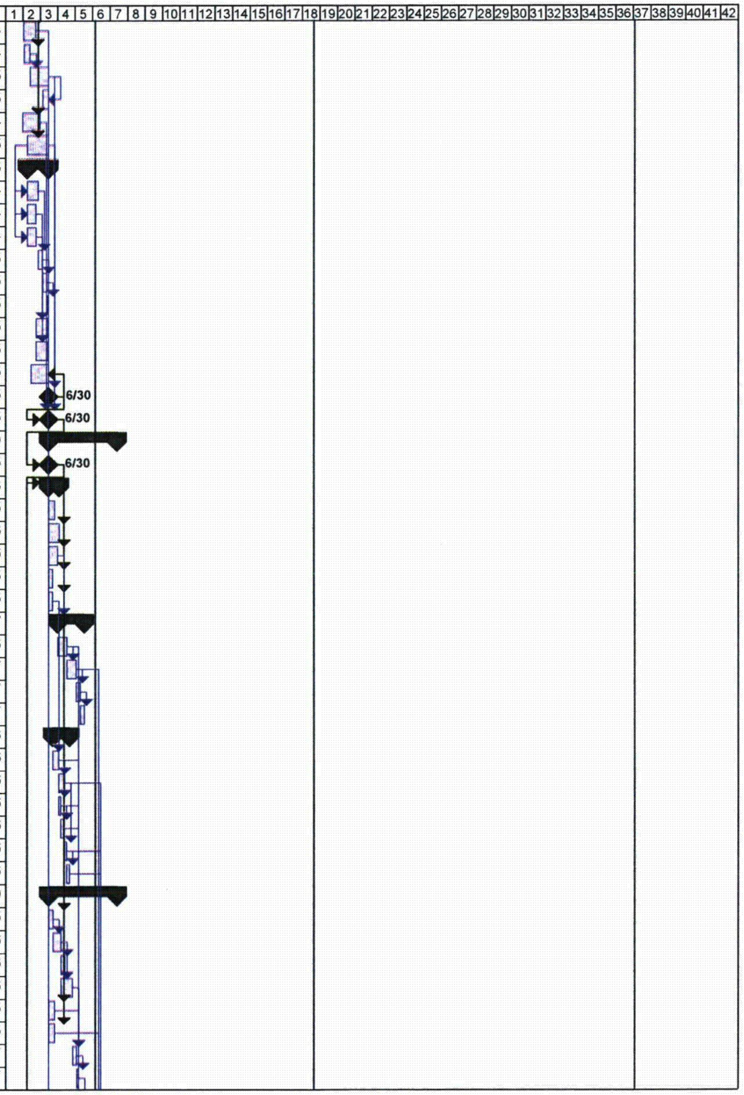
SONGS 2 & 3
Detailed Project Schedule
Prompt DECON, DOE Repository Opens 2024

ID	Task Name	Duration	Start	Finish
48	SNF D&D Pd 2 Begins	0 days	05/06/2050	05/06/2050
49	Install GARDIAN Bulk Assay System	30 days	05/07/2050	06/17/2050
50	Decon AHSMs	90 days	05/07/2050	09/09/2050
51	Final Status Survey of ISFSI	120 days	05/28/2050	11/11/2050
52	Clean Demolition of ISFSI AHSMs and Pad	145 days	09/10/2050	03/31/2051
53	Clean Demolition of ISFSI Support Structures	120 days	10/15/2050	03/31/2051
54	Restore ISFSI Site	55 days	04/01/2051	06/16/2051
55	Preparation of Final Report on Decommissioning and NRC Review	60 days	06/17/2051	09/08/2051
56	SNF D&D Pd 2 Ends - License Termination Complete	0 days	09/08/2051	09/08/2051
57	Post-Shutdown Spent Fuel Management Complete	0 days	09/08/2051	09/08/2051
58	**Part 50 License Termination**	**5102 days**	**06/07/2013**	**12/24/2032**
59	Announcement of Cessation of Operations (June 7, 2013)	0 days	06/07/2013	06/07/2013
60	**Decon Pd 1 - Transition to Decommissioning**	**149 days**	**06/07/2013**	**12/31/2013**
61	Decon Pd 1 Begins	0 days	06/07/2013	06/07/2013
62	Certification of Permanent Cessation Submitted to NRC (June 12, 2013)	0 days	06/07/2013	06/07/2013
63	Defuel Unit 3 Reactor	15 days	06/07/2013	06/27/2013
64	Defuel Unit 2 Reactor	15 days	06/07/2013	06/27/2013
65	Notification of Permanent Fuel Removal (July 23, 2013)	0 days	06/27/2013	06/27/2013
66	Disposition of Legacy Wastes	60 days	07/19/2013	10/10/2013
67	Decon Pd 1 Ends	0 days	01/01/2014	01/01/2014
68	**Decon Pd 2 - Decommissioning Planning and Site Modifications**	**389 days**	**1/1/2014**	**06/30/2015**
69	Decon Pd 2 Begins	0 wks	01/01/2014	01/01/2014
70	**Preparation of Decommissioning License Documents**	**340 days**	**01/02/2014**	**04/22/2015**
71	Develop Certified Fuel Handler Program	340 days	01/02/2014	04/22/2015
72	Prepare Post-Shutdown QA Plan	340 days	01/02/2014	04/22/2015
73	Prepare Post-Shutdown Security Plan	340 days	01/02/2014	04/22/2015
74	Prepare Post-Shutdown Fire Protection Plan	340 days	01/02/2014	04/22/2015
75	Prepare Defueled Radiation Protection Manual	340 days	01/02/2014	04/22/2015
76	Prepare Preliminary Defueled Technical Specifications	63 days	01/02/2014	03/29/2014
77	**NRC Deliverables**	**364 days**	**01/02/2014**	**05/26/2015**
78	Prepare Defueled Safety Analysis Report (DSAR)	311 days	01/02/2014	03/12/2015
79	Submit DSAR to NRC	0 days	03/12/2015	03/12/2015
80	Implement Technical Specification Modifications	30 days	03/13/2015	04/23/2015
81	Prepare Post-Shutdown Emergency Preparedness Plan	304 days	01/02/2014	03/03/2015
82	Submit Emergency Plan to NRC	0 days	03/03/2015	03/03/2015
83	NRC Review of Emergency Plan	60 days	03/04/2015	05/26/2015
84	Prepare Post-Shutdown Decommissioning Activities Report (PSDAR)	121 days	01/02/2014	06/19/2014
85	Submit PSDAR to NRC	0 days	06/19/2014	06/19/2014
86	NRC Review of PSDAR	90 days	06/20/2014	10/23/2014
87	Public Meeting on PSDAR	30 days	08/01/2014	09/11/2014
88	Prepare Decommissioning Cost Estimate (DCE)	160 days	01/02/2014	08/13/2014
89	Submit DCE to NRC	0 days	08/13/2014	08/13/2014
90	NRC Review of Decommissioning Cost Estimate	90 days	08/14/2014	12/17/2014
91	Commencement of Major Decommisisoning Activities Allowable	0 days	10/23/2014	10/23/2014
92	Respond to NRC quesitons on PSDAR	220 days	06/20/2014	04/23/2015
93	Disposition of Legacy Wastes	220 days	01/02/2014	11/05/2014
94	Contract Award for Historic Site Assessment and Site Characterization	0 wks	01/16/2014	01/16/2014



SONGS 2 & 3
Detailed Project Schedule
Prompt DECON, DOE Repository Opens 2024

ID	Task Name	Duration	Start	Finish
95	Perform Historic Site Assessment and Site Characterization	180 days	01/17/2014	09/25/2014
96	Planning and Design For Cold and Dark	90 days	02/01/2014	06/06/2014
97	Implement Cold and Dark (Repower Site)	275 days	06/07/2014	06/26/2015
98	Install 12kV Service Line to Power Temporary Power Ring	90 days	02/21/2015	06/26/2015
99	Drain and De-Energize Non-Essential Systems (DEC Process)	260 days	01/02/2014	12/31/2014
100	Select Decommissioning General Contractor (DGC)	318 days	04/11/2014	06/30/2015
101	**Spent Fuel Pool Isolation**	**318 days**	**04/11/2014**	**06/30/2015**
102	Design Spent Fuel Pool Support System Modifications	160 days	04/11/2014	11/20/2014
103	Design Control Room Relocation	125 days	04/11/2014	10/02/2014
104	Design Spent Fuel Security System Modifications	130 days	04/11/2014	10/09/2014
105	Install Spent Fuel Pool System Modifications - Unit 2	66 days	11/21/2014	02/20/2015
106	Install Spent Fuel Pool System Modifications - Unit 3	66 days	02/21/2015	05/23/2015
107	Spent Fuel Pool Island System Training	10 days	05/26/2015	06/06/2015
108	Implement Control Room Modifications	185 days	10/03/2014	06/18/2015
109	Implement Spent Fuel Pool Security Modifications	180 days	10/10/2014	06/18/2015
110	Transition Project Modifications	262 days	06/28/2014	06/30/2015
111	DGC Contract Award	0 days	06/30/2015	06/30/2015
112	Decon Pd 2 Ends	0 wks	06/30/2015	06/30/2015
113	**Decon Pd 3 - Decommissioning Preparations and Reactor Internals Segmenta**	**1024 days**	**06/30/2015**	**06/01/2019**
114	Decon Pd 3 Begins	0 days	06/30/2015	06/30/2015
115	**DGC Mobilization and Planning**	**160 days**	**07/01/2015**	**02/09/2016**
116	Prepare Integrated Work Sequence and Schedule for Decommissioning	90 days	07/01/2015	11/03/2015
117	Prepare Detailed Work Procedures for Decommissioning	160 days	07/01/2015	02/09/2016
118	Planning and Design of Primary System Decontamination	135 days	07/01/2015	01/05/2016
119	Planning and Design of Infrastructure Improvements	60 days	07/01/2015	09/22/2015
120	Design Containment Access Modifications	60 days	07/01/2015	09/22/2015
121	**System Decon**	**400 days**	**01/06/2016**	**07/18/2017**
122	Perform Primary System Decon- Unit 2	140 days	01/06/2016	07/19/2016
123	Perform Primary System Decon- Unit 3	140 days	07/20/2016	01/31/2017
124	Hot Spot Decontamination - Unit 2	60 days	02/01/2017	04/25/2017
125	Hot Spot Decontamination - Unit 3	60 days	04/26/2017	07/18/2017
126	**Rx Internals Removal Preparations**	**255 days**	**09/23/2015**	**09/13/2016**
127	Modify Containment Access- Unit 2	90 days	09/23/2015	01/26/2016
128	Modify Containment Access- Unit 3	90 days	01/27/2016	05/31/2016
129	Remove and Dispose of Missle Shields - Unit 2	30 days	01/27/2016	03/08/2016
130	Remove and Dispose of Reactor Head - Unit 2	45 days	03/09/2016	05/10/2016
131	Remove and Dispose of Missle Shields - Unit 3	30 days	06/01/2016	07/12/2016
132	Remove and Dispose of Reactor Head - Unit 3	45 days	07/13/2016	09/13/2016
133	**Reactor Internals Segmentation Planning and Implementation**	**1020 days**	**07/01/2015**	**05/28/2019**
134	Finalize Residual Radiation Inventory (Rx Vessel & Internals)	65 days	07/01/2015	09/29/2015
135	Prepare Activity Specification for Rx Vessel and Internals Segmentation	120 days	09/30/2015	03/15/2016
136	Select Shipping Casks and Obtain Shipping Permits	60 days	03/16/2016	06/07/2016
137	Design, Specify, and Procure Special Items and Materials	175 days	03/16/2016	11/15/2016
138	Purchase Dry Storage Modules for GTCC Waste - Unit 2	90 days	07/01/2015	11/03/2015
139	Purchase Dry Storage Modules for GTCC Waste - Unit 3	90 days	07/01/2015	11/03/2015
140	Test Special Cutting and Handling Equipment and Train Operators	60 days	11/16/2016	02/07/2017
141	Finalize Internals and Vessel Segmenting Details - Unit 2	30 days	02/08/2017	03/21/2017



SONGS 2 & 3
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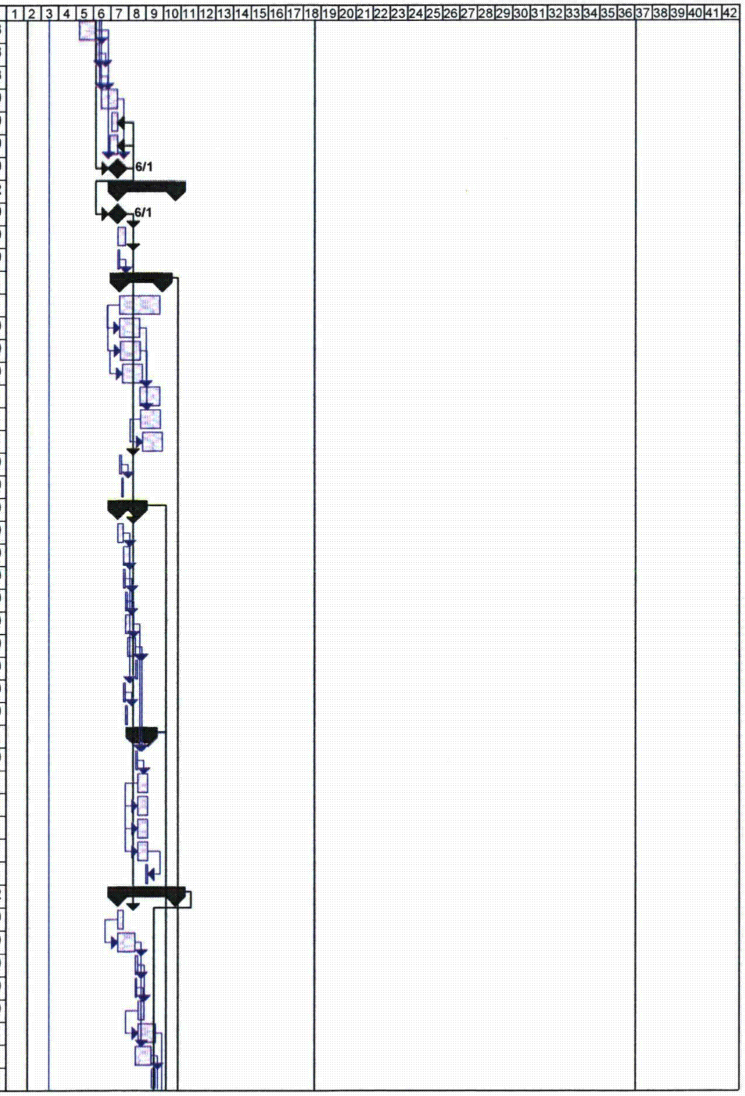
ID	Task Name	Duration	Start	Finish
142	Segment, Package and Dispose of Reactor Internals - Unit 2	240 days	03/22/2017	02/20/2018
143	Transfer Internals Segmentation Equipment to Unit 3	60 days	02/21/2018	05/15/2018
144	Finalize Internals and Vessel Segmenting Details - Unit 3	30 days	05/16/2018	06/26/2018
145	Segment, Package and Dispose of Reactor Internals - Unit 3	240 days	06/27/2018	05/28/2019
146	Construct new change rooms, hot laundry, in-plant laydown areas	90 days	01/29/2019	06/01/2019
147	Procure Non-Engineered Standard Equipment	120 days	12/18/2018	06/01/2019
148	Decon Pd 3 Ends	0 wks	06/01/2019	06/01/2019
149	**Decon Pd 4 - Plant Systems and Large Component Removal**	**865 days**	**06/01/2019**	**09/24/2022**
150	Decon Pd 4 Begins	0 days	06/01/2019	06/01/2019
151	Upgrade Rail Spur	120 days	06/04/2019	11/16/2019
152	Install GARDIAN Bulk Assay System	30 days	06/04/2019	07/13/2019
153	**Non Essential System Removal**	**640 days**	**07/16/2019**	**12/25/2021**
154	Scaffolding for Non-Essential System Removal	120 wks	07/16/2019	10/30/2021
155	Asbestos Abatement for Non-Essential Systems Removal - Unit 2	60 wks	07/16/2019	09/05/2020
156	Lead Abatement for Non-Essential Systems Removal - Unit 2	60 wks	07/30/2019	09/19/2020
157	Remove, Package and Dispose of Non-Essential Systems - Unit 2	60 wks	09/10/2019	10/31/2020
158	Asbestos Abatement for Non-Essential Systems - Unit 3	60 wks	09/08/2020	10/30/2021
159	Lead Abatement for Non-Essential Systems - Unit 3	60 wks	09/22/2020	11/13/2021
160	Remove, Package and Dispose of Non-Essential Systems - Unit 3	60 wks	11/03/2020	12/25/2021
161	Remove Underground Diesel Tank - Unit 2	30 days	07/16/2019	08/24/2019
162	Remove Underground Diesel Tank - Unit 3	30 days	08/27/2019	10/05/2019
163	**Fuel Pool Closure**	**300 days**	**06/04/2019**	**07/25/2020**
164	Remove and Dispose of Spent Fuel Storage Racks - Unit 2	90 days	06/04/2019	10/05/2019
165	Remove and Dispose of Spent Fuel Storage Racks - Unit 3	90 days	10/08/2019	02/08/2020
166	Remove and Dispose of Legacy Class B and C Waste - Unit 2	30 days	10/08/2019	11/16/2019
167	Remove and Dispose of Legacy Class B and C Waste - Unit 3	30 days	11/19/2019	12/28/2019
168	Drain Spent Fuel Pool and Process Liquid Waste - Unit 2	24 wks	11/19/2019	05/02/2020
169	Drain Spent Fuel Pool and Process Liquid Waste - Unit 3	24 wks	12/31/2019	06/13/2020
170	Segment, Package and Dispose of Spent Fuel Pool Island Equipment	30 days	06/16/2020	07/25/2020
171	Segment and Dispose of Fuel Pool Bridge Crane - Unit 2	30 days	10/08/2019	11/16/2019
172	Segment and Dispose of Fuel Pool Bridge Crane - Unit 3	30 days	11/19/2019	12/28/2019
173	**Essential Systems Removal**	**180 days**	**06/16/2020**	**02/20/2021**
174	Flush and Drain Essential Systems Following Fuel Pool Closure	30 days	06/16/2020	07/25/2020
175	Scaffolding for Essential System Removal	30 wks	07/28/2020	02/20/2021
176	Asbestos Abatement for Essential Systems	30 wks	07/28/2020	02/20/2021
177	Lead Abatement for Essential Systems Removal	30 wks	07/28/2020	02/20/2021
178	Remove, Package and Dispose of Essential Systems	30 wks	07/28/2020	02/20/2021
179	Removal and Disposal of Spent Resins, Filter Media and Tank Sludge	30 days	01/12/2021	02/20/2021
180	**Large Component Removal**	**865 days**	**06/04/2019**	**09/24/2022**
181	Reactor Vessel Insulation Removal and Disposal - Unit 2	90 days	06/04/2019	10/05/2019
182	Segment, Package and Dispose of Reactor Pressure Vessel - Unit 2	260 days	06/04/2019	05/30/2020
183	Transfer Rx Vessel Segmentation Equipment to Unit 3	45 days	06/02/2020	08/01/2020
184	Procure Replacement Non-Engineered Standard Equipment	30 days	06/02/2020	07/11/2020
185	Reactor Vessel Insulation Removal and Disposal - Unit 3	90 days	08/04/2020	12/05/2020
186	Segment, Package and Dispose of Reactor Pressure Vessel - Unit 3	260 days	08/04/2020	07/31/2021
187	Remove and Dispose of Steam Generators - Unit 2	240 days	06/02/2020	05/01/2021
188	Remove and Dispose of Pressurizer - Unit 2	60 days	05/04/2021	07/24/2021



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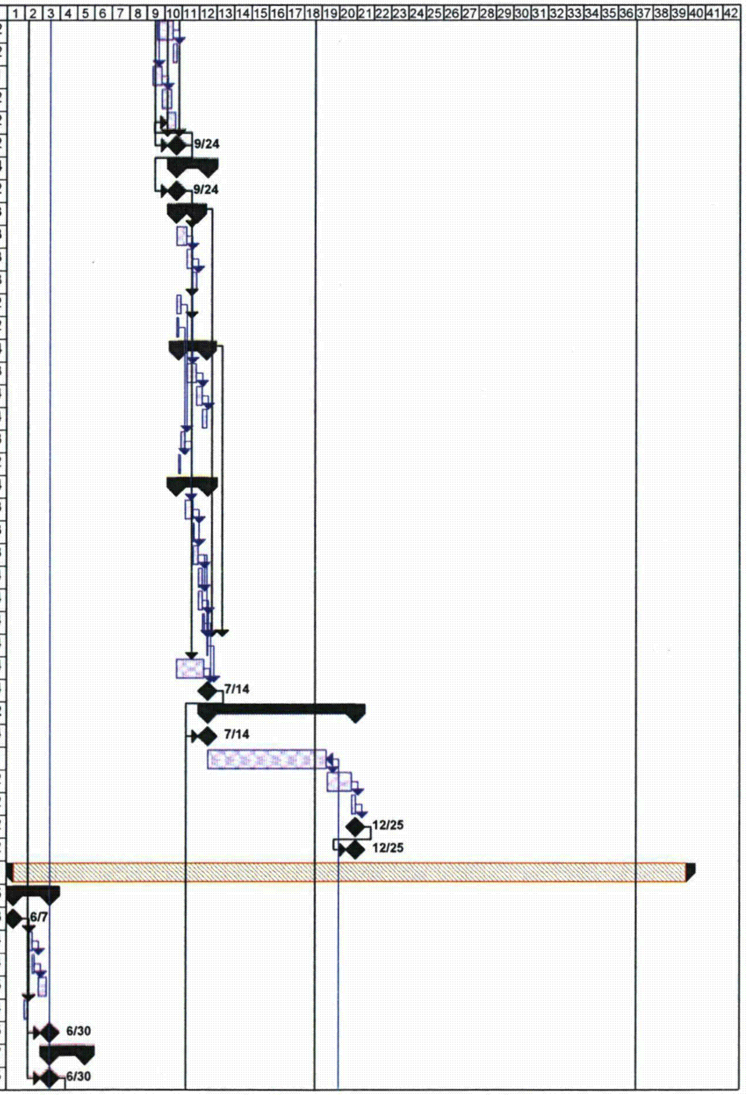
ID	Task Name	Duration	Start	Finish
189	Remove and Dispose of Steam Generators - Unit 3	240 days	08/03/2021	07/02/2022
190	Remove and Dispose of Pressurizer - Unit 3	60 days	07/05/2022	09/24/2022
191	Remove and Dispose of Turbine Gantry Crane - Unit 2	140 days	05/04/2021	11/13/2021
192	Remove and Dispose of Turbine Gantry Crane - Unit 3	140 days	11/16/2021	05/28/2022
193	Prepare License Termination Plan	26 wks	03/01/2022	08/27/2022
194	Decon Pd 4 Ends	0 days	09/24/2022	09/24/2022
195	**Decon Pd 5 - Building Decontamination**	**470 days**	**09/24/2022**	**07/13/2024**
196	Decon Pd 5 Begins	0 days	09/24/2022	09/24/2022
197	**Unit 3**	**305 days**	**09/27/2022**	**11/25/2023**
198	Decon Containment Building - Unit 3	150 days	09/27/2022	04/22/2023
199	Decon Penetration Building - Unit 3	85 days	04/25/2023	08/19/2023
200	Decon Safety Equipment and MSIV Building - Unit 3	70 days	08/22/2023	11/25/2023
201	Decon Fuel Handling Building - Unit 3	65 days	09/27/2022	12/24/2022
202	Decon Turbine Building - Unit 3	30 days	09/27/2022	11/05/2022
203	**Unit 2**	**425 days**	**11/08/2022**	**06/22/2024**
204	Decon Containment Building - Unit 2	150 days	04/25/2023	11/18/2023
205	Decon Penetration Building - Unit 2	85 days	11/21/2023	03/16/2024
206	Decon Safety Equipment and MSIV Building - Unit 2	70 days	03/19/2024	06/22/2024
207	Decon Fuel Handling Building - Unit 2	65 days	12/27/2022	03/25/2023
208	Decon Turbine Building - Unit 2	30 days	11/08/2022	12/17/2022
209	**Common**	**470 days**	**09/27/2022**	**07/13/2024**
210	Decon Auxiliary Radwaste Building - Common	120 days	03/28/2023	09/09/2023
211	Decon Auxiliary Control Building - Common	20 days	09/12/2023	10/07/2023
212	Decon Condensate Area and Tunnels - Units 2 and 3	80 days	09/12/2023	12/30/2023
213	Excavate, Remove and Dispose of Yard Area Drains	60 days	01/02/2024	03/23/2024
214	Remove and Dispose of Contaminated Sumps, Trenches and Pavement	60 days	01/02/2024	03/23/2024
215	Remove and Dispose of Radiologically Contaminated Soil	30 days	03/26/2024	05/04/2024
216	Dispose of Contaminated Decon Equipment and Tooling	15 days	06/25/2024	07/13/2024
217	Radiological Survey of Structures During Decon	410 days	09/27/2022	04/20/2024
218	Decon Pd 5 Ends	0 days	07/13/2024	07/13/2024
219	**Decon Pd 6 - License Termination During Demolition**	**2206 days**	**07/13/2024**	**12/24/2032**
220	Decon Pd 6 Begins	0 days	07/13/2024	07/13/2024
221	Final Status Survey	1771 days	07/13/2024	04/25/2031
222	ORISE Verification and NRC Approval	18 mons	05/17/2031	10/01/2032
223	Prepare Final Report of Dismantling Program	60 days	10/02/2032	12/24/2032
224	Decon Complete - Partial License Termination	0 days	12/24/2032	12/24/2032
225	Decon Pd 6 Ends	0 days	12/24/2032	12/24/2032
226	**Site Restoration**	**10052 days**	**06/07/2013**	**12/15/2051**
227	**SR Pd 1 - Transition to Site Restoration**	**538 days**	**06/07/2013**	**06/30/2015**
228	SR Pd 1 Begins	0 days	06/07/2013	06/07/2013
229	Mesa Site Phase I and II Site Assessment	60 days	04/11/2014	07/03/2014
230	Disposition Hazardous Waste from Mesa Site	30 days	07/04/2014	08/14/2014
231	Mesa Site Characterization Survey	120 days	11/07/2014	04/23/2015
232	Fuel Cancellation Expense	60 days	01/21/2014	04/12/2014
233	SR Pd 1 Ends	0 days	06/30/2015	06/30/2015
234	**SR Pd 2 - Building Demolition During Decommissioning**	**530 days**	**06/30/2015**	**07/11/2017**
235	SR Pd 2 Begins	0 days	06/30/2015	06/30/2015



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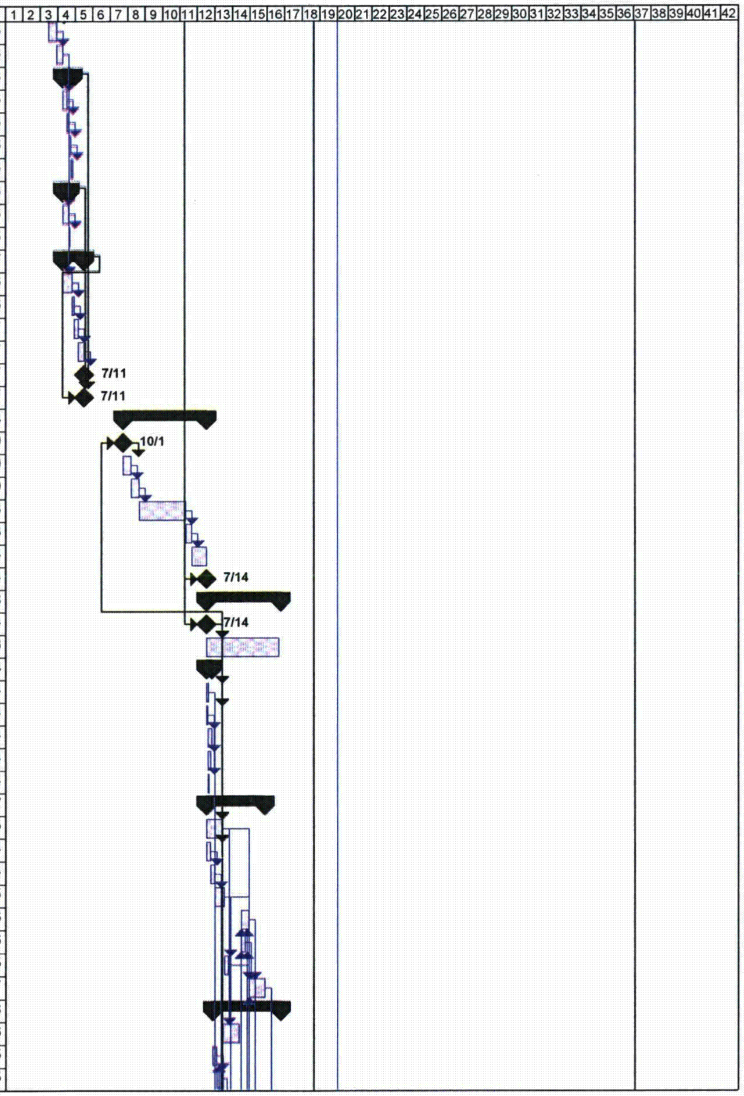
ID	Task Name	Duration	Start	Finish
236	Prepare Site Restoration Demolition Plan and Schedule	120 days	07/01/2015	12/15/2015
237	Obtain Required Permits For Mesa, South Access and South Yard	90 days	12/16/2015	04/19/2016
238	**South Access for Decommissioning**	**150 days**	**04/20/2016**	**11/15/2016**
239	Demolish Service Building (K-10, 20, 30)	60 days	04/20/2016	07/12/2016
240	Demolish South Security Processing Facility (K-70)	30 days	07/13/2016	08/23/2016
241	Demolish Staging Warehouse	30 days	08/24/2016	10/04/2016
242	Demolish Administration Building (K-40/50)	30 days	10/05/2016	11/15/2016
243	**South Yard Facility**	**105 days**	**04/20/2016**	**09/13/2016**
244	Demolish South Yard Area Buildings T-10, 20, 60 and Haz Mat.	90 days	04/20/2016	08/23/2016
245	Demolish REMS Staging Pad	15 days	08/24/2016	09/13/2016
246	**Mesa**	**320 days**	**04/20/2016**	**07/11/2017**
247	Demolish Mesa Buildings	140 days	04/20/2016	11/01/2016
248	Remove Underground Fuel Storage Tanks	30 days	11/02/2016	12/13/2016
249	Demolish Mesa Roads and Parking Lots	60 days	12/14/2016	03/07/2017
250	Finish Grading and Re-vegetate Mesa Site	90 days	03/08/2017	07/11/2017
251	Mesa Area Cleared for Easement Termination	0 days	07/11/2017	07/11/2017
252	SR Pd 2 Ends	0 days	07/11/2017	07/11/2017
253	**SR Pd 3 - Subsurface Demolition Engineering and Permitting**	**1250 days**	**10/01/2019**	**07/13/2024**
254	SR Pd 3 Begins	0 days	10/01/2019	10/01/2019
255	Hydrogeologic Investigation and Outfall Conduit Survey	120 days	10/01/2019	03/14/2020
256	Subsurface Structure Removal Engineering Planning and Design	120 days	03/17/2020	08/29/2020
257	Environmental Impacts Analyses for Lease Termination Activities	700 days	09/01/2020	05/06/2023
258	Final Site Grading and Shoreline Protection Engineering Planning and Design	90 days	05/09/2023	09/09/2023
259	Obtain Required Permits and Approvals	220 days	09/12/2023	07/13/2024
260	SR Pd 3 Ends	0 days	07/13/2024	07/13/2024
261	**SR Pd 4 - Building Demolition to 3 Feet Below Grade**	**1110 days**	**07/13/2024**	**10/14/2028**
262	SR Pd 4 Begins	0 days	07/13/2024	07/13/2024
263	Procure Building Demolition Equipment	1080 days	07/16/2024	09/02/2028
264	**Demolition Preparations**	**80 days**	**07/16/2024**	**11/02/2024**
265	Install Temporary Structures	30 days	07/16/2024	08/24/2024
266	Install Erosion and Sediment Controls	20 days	07/16/2024	08/10/2024
267	Remove Cathodic Protection Trench	60 days	08/13/2024	11/02/2024
268	Remove Protected Area Security Fencing	45 days	08/13/2024	10/12/2024
269	Remove Protected Area Pavement	20 days	08/13/2024	09/07/2024
270	**Unit 3**	**870 days**	**07/16/2024**	**11/13/2027**
271	Detension and Remove Unit 3 Containment Builidng Tendons	240 days	07/16/2024	06/14/2025
272	Demolish Diesel Generator Building - Unit 3	60 days	07/16/2024	10/05/2024
273	Demolish Condensate Building and Transformer Pads - Unit 3	60 days	10/08/2024	12/28/2024
274	Demolish Full Flow Area and Turbine Building - Unit 3	140 days	12/31/2024	07/12/2025
275	Demolish Unit 3 Fuel Handling Building to 3-Feet Below Grade	120 days	06/30/2026	12/12/2026
276	Demolish Penetration Building - Unit 3	60 days	06/30/2026	09/19/2026
277	Demolish Safety Equipmentand MSIV Building - Unit 3	60 days	07/15/2025	10/04/2025
278	Demolish Unit 3 Containment Building to 3-Feet Below Grade	240 days	12/15/2026	11/13/2027
279	**Unit 2**	**1020 days**	**11/19/2024**	**10/14/2028**
280	Detension and Remove Unit 2 Containment Builidng Tendons	240 days	06/17/2025	05/16/2026
281	Demolish Diesel Generator Building - Unit 2	60 days	11/19/2024	02/08/2025
282	Demolish Condensate Building and Transformer Pads - Unit 2	60 days	02/11/2025	05/03/2025



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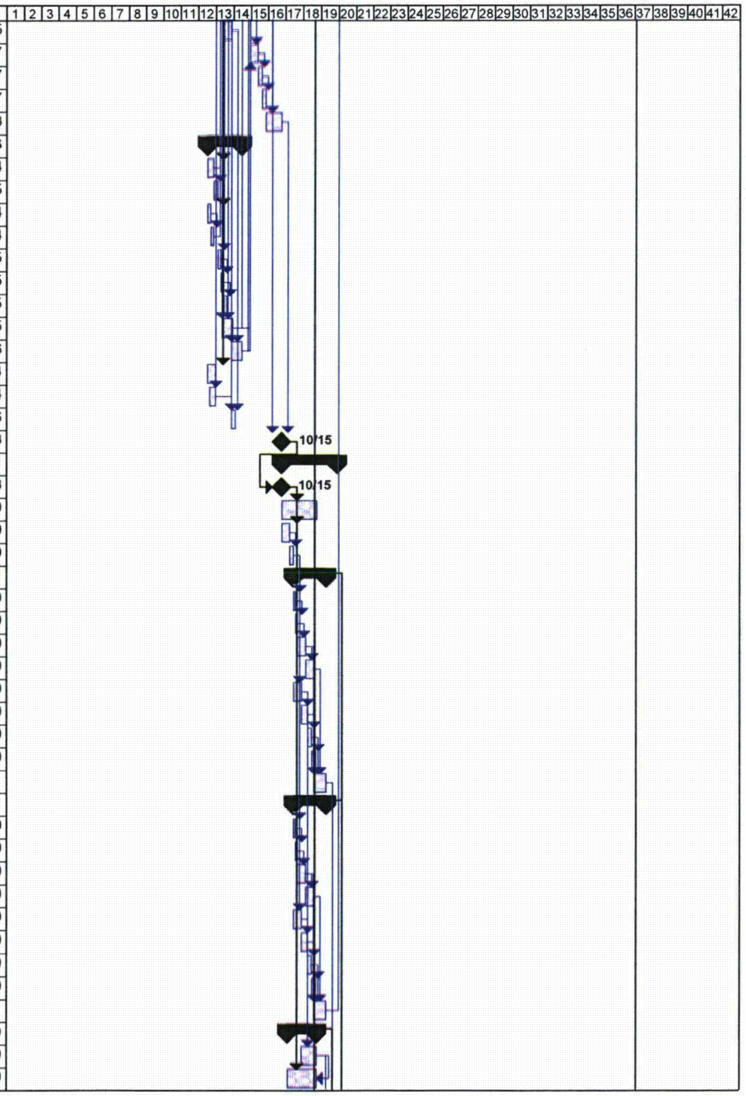
ID	Task Name	Duration	Start	Finish
283	Demolish Full Flow Area and Turbine Building - Unit 2	140 days	05/06/2025	11/15/2025
284	Demolish Unit 2 Fuel Handling Building to 3-Feet Below Grade	120 days	12/15/2026	05/29/2027
285	Demolish Penetration Building - Unit 2	60 days	06/01/2027	08/21/2027
286	Demolish Safety Equipment and MSIV Building - Unit 2	60 days	08/24/2027	11/13/2027
287	Demolish Unit 2 Containment Building to 3-Feet Below Grade	240 days	11/16/2027	10/14/2028
288	**Common**	**510 days**	**07/16/2024**	**06/27/2026**
289	Demolish AWS Building	90 days	07/16/2024	11/16/2024
290	Demolish Building L-50	60 days	11/19/2024	02/08/2025
291	Demolish Building B-64/B-65	45 days	07/16/2024	09/14/2024
292	Demolish Building B-62/B-63	45 days	09/17/2024	11/16/2024
293	Demolish Outage Control Center	45 days	02/11/2025	04/12/2025
294	Demolish Building B-49/B-50	45 days	04/15/2025	06/14/2025
295	Demolish Building B-43/B-44	45 days	06/17/2025	08/16/2025
296	Demolish Auxiliary Radwaste Building - Common	160 days	05/06/2025	12/13/2025
297	Demolish Auxiliary Control Building - Common	160 days	11/18/2025	06/27/2026
298	Remove Systems and Demolish Make-Up Demineralizer Structures	120 days	07/16/2024	12/28/2024
299	Install Concrete Plugs in Intake and Discharge Structures	90 days	08/27/2024	12/28/2024
300	Demolish Intake and Discharge Structures to 3-Feet Below Grade	60 days	11/18/2025	02/07/2026
301	SR Pd 4 Ends	0 days	10/14/2028	10/14/2028
302	**SR Pd 5 - Subgrade Structure Removal Below -3 Feet**	**820 days**	**10/14/2028**	**12/05/2031**
303	SR Pd 5 Begins	0 days	10/14/2028	10/14/2028
304	Procure Subsurface Structure Demolition Equipment	520 days	10/17/2028	10/11/2030
305	Install Sheet Piling and Excavation Shoring	120 days	10/17/2028	03/31/2029
306	Install Dewatering System and Effluent Treatment and Discharge Controls	60 days	04/01/2029	06/22/2029
307	**Unit 3 Subsurface Structures**	**480 days**	**06/23/2029**	**04/25/2031**
308	Demolish and Backfill Unit 3 Condensate Storage Area Below -3 Feet	30 days	06/23/2029	08/03/2029
309	Demolish and Backfill Unit 3 Diesel Generator Builidng Below -3 Feet	30 days	08/04/2029	09/14/2029
310	Demolish and Backfill Unit 3 Fuel Handling Building Below -3 Feet	120 days	09/15/2029	03/01/2030
311	Demolish and Backfill Unit 3 Radwaste Building Below -3 Feet	120 days	03/02/2030	08/16/2030
312	Demolish and Backfill Unit 3 Turbine Building Structure Below 9 Ft Elevatior	120 days	06/23/2029	12/07/2029
313	Demolish and Backfill Unit 3 Safety Equipment Building Below -3 Feet	90 days	12/08/2029	04/12/2030
314	Demolish and Backfill Unit 3 Penetration Area Below -3 Feet	60 days	04/13/2030	07/05/2030
315	Demolish and Backfill Unit 3 Full Flow Building Below -3 Feet	60 days	07/06/2030	09/27/2030
316	Demolish and Backfill Unit 3 Containment Building Below -3 Feet	180 days	08/17/2030	04/25/2031
317	**Unit 2 Subsurface Structures**	**480 days**	**06/23/2029**	**04/25/2031**
318	Demolish and Backfill Unit 2 Condensate Storage Area Below -3 Feet	30 days	06/23/2029	08/03/2029
319	Demolish and Backfill Unit 2 Diesel Generator Builidng Below -3 Feet	30 days	08/04/2029	09/14/2029
320	Demolish and Backfill Unit 2 Fuel Handling Building Below -3 Feet	120 days	09/15/2029	03/01/2030
321	Demolish and Backfill Unit 2 Radwaste Building Below -3 Feet	120 days	03/02/2030	08/16/2030
322	Demolish and Backfill Unit 2 Turbine Building Structure Below 9 Ft Elevatior	120 days	06/23/2029	12/07/2029
323	Demolish and Backfill Unit 2 Safety Equipment Building Below -3 Feet	90 days	12/08/2029	04/12/2030
324	Demolish and Backfill Unit 2 Penetration Area Below -3 Feet	60 days	04/13/2030	07/05/2030
325	Demolish and Backfill Unit 2 Full Flow Building Below -3 Feet	60 days	07/06/2030	09/27/2030
326	Demolish and Backfill Unit 2 Containment Building Below -3 Feet	180 days	08/17/2030	04/25/2031
327	**Common Subgrade Structures**	**432 days**	**02/16/2029**	**10/11/2030**
328	Demolish and Backfill Intake Structure Inside Seawall Below -3 Feet	220 days	12/08/2029	10/11/2030
329	Remove Off Shore Intake and Outfall Conduits	432 days	02/16/2029	10/11/2030



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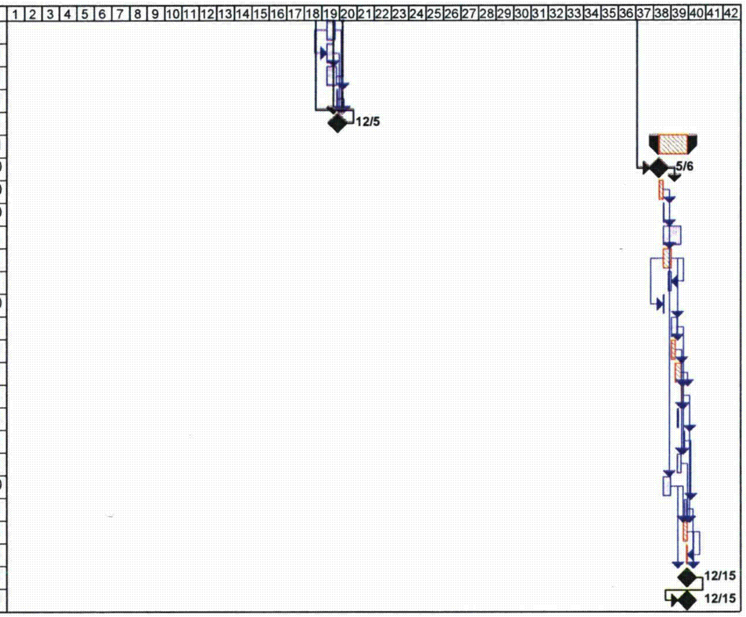
ID	Task Name	Duration	Start	Finish
330	Remove Sheet Piling and Excavation Shoring	120 days	04/26/2031	10/10/2031
331	Remove Dewatering System and Effluent Treatment	90 days	04/26/2031	08/29/2031
332	Finish Grading and Re-Vegetate Site	140 days	04/26/2031	11/07/2031
333	Remove Temporary Structures	20 days	11/08/2031	12/05/2031
334	SR Pd 5 Ends	0 days	12/05/2031	12/05/2031
335	**SR Pd 6 - Final Site Restoration and Lease Termination**	**420 days**	**05/06/2050**	**12/15/2051**
336	SR Pd 6 Begins	0 days	05/06/2050	05/06/2050
337	Obtain Required Permits and Approvals	60 days	05/07/2050	07/29/2050
338	Install Temporary Structures	10 days	07/30/2050	08/12/2050
339	Procure Site Restoration Equipment	265 days	07/30/2050	08/04/2051
340	Install Temporary Seawall or Coffer Dam	120 days	07/30/2050	01/13/2051
341	Install Dewatering System and Effluent Treatment and Discharge Controls	45 days	11/12/2050	01/13/2051
342	Remove and Stockpile Existing Seawall Erosion Protection	10 days	07/30/2050	08/12/2050
343	Remove Unit 2 and 3 Seawall and Pedestrian Walkway	90 days	01/14/2051	05/19/2051
344	Remove Remaining Intake and Outfall Box Culvert	60 days	01/14/2051	04/07/2051
345	Remove Temporary Seawall or Coffer Dam	90 days	04/08/2051	08/11/2051
346	Backfill and Compaction of Excavation	30 days	08/12/2051	09/22/2051
347	Remove Dewatering System and Effluent Treatment	20 days	05/20/2051	06/16/2051
348	Install Shoreline Erosion Control and Restoration Features	20 days	09/23/2051	10/20/2051
349	Remove Railroad Tracks, Rails and Ballast	60 days	05/20/2051	08/11/2051
350	Remove Gunite Slope Protection	110 days	07/30/2050	12/30/2050
351	Remove Access Roads and Parking Lots	30 days	10/21/2051	12/01/2051
352	Finish Grading and Re-Vegetate Site	60 days	09/23/2051	12/15/2051
353	Remove Temporary Structures	10 days	12/02/2051	12/15/2051
354	SR Pd 6 Ends	0 days	12/15/2051	12/15/2051
355	Final Easement Termination	0 days	12/15/2051	12/15/2051



Appendix D

Detailed Cost Table

Table 1
SONGS Prompt DECON Base Case, 2024 DOE Acceptance, Dry Storage

		License Status	POL	Unit 2 Shut Down:	6/7/2013
Decommissioning Alternative	DECON	Fuel Pool Systems	Modified	Unit 3 Shut Down:	6/7/2013
Spent Fuel Alternative	Dry	Repository Opening Date:	1/1/2024		

2014 Dollars in Thousands

No	Item Description	Labor	Equipment	Disposal	Other	Contingency	Total
A. License Termination							
Decon Pd 1	**Transition to Decommissioning**						
Distributed							
1.05	Disposition of Legacy Wastes	$0	$0	$9,153	$735	$0	$9,888
Distributed	**Subtotal**	**$0**	**$0**	**$9,153**	**$735**	**$0**	**$9,888**
Undistributed							
1.01	Utility Staff	$30,049	$0	$0	$0	$0	$30,049
1.05	Insurance	$0	$0	$0	$5,352	$0	$5,352
1.07	NRC Decommissioning Fees	$0	$0	$0	$1,349	$0	$1,349
1.08	Materials and Services	$0	$0	$0	$1,007	$0	$1,007
1.10	Energy	$0	$0	$0	$2,422	$0	$2,422
1.17	Association Fees and Expenses	$0	$0	$0	$315	$0	$315
1.18	Utilities (Water, gas, phone)	$0	$0	$0	$840	$0	$840
1.20	Non-Process Computers	$0	$0	$0	$224	$0	$224
1.21	Telecommunications	$0	$0	$0	$41	$0	$41
1.22	Personal Computers	$0	$0	$0	$9	$0	$9
1.24	Environmental Permits and Fees	$0	$0	$0	$818	$0	$818
Undistributed	**Subtotal**	**$30,049**	**$0**	**$0**	**$12,378**	**$0**	**$42,426**
Decon Pd 1	**Subtotal**	**$30,049**	**$0**	**$9,153**	**$13,113**	**$0**	**$52,315**

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2014 Dollars in Thousands

No	Item Description	Labor	Equipment	Disposal	Other	Contingency	Total
Decon Pd 2	**Decommissioning Planning and Site Modifications**						
Distributed							
2.01	Develop Certified Fuel Handler Program	$143	$1	$0	$0	$36	$180
2.02	Prepare Post-Shutdown QA Plan	$427	$1	$0	$0	$107	$535
2.03	Prepare Post-Shutdown Security Plan	$427	$1	$0	$0	$107	$535
2.04	Prepare Post-Shutdown Fire Protection Plan	$427	$1	$0	$0	$107	$535
2.05	Prepare Defueled Radiation Protection Manual	$427	$1	$0	$0	$107	$535
2.06	Prepare Preliminary Defueled Technical Specifications	$0	$0	$0	$135	$34	$169
2.07	Prepare Defueled Safety Analysis Report (DSAR)	$1,279	$5	$0	$0	$321	$1,605
2.08	Implement Technical Specification Modifications	$1,332	$5	$0	$0	$334	$1,671
2.09	Prepare Post-Shutdown Emergency Preparedness Plan	$634	$1	$0	$0	$159	$793
2.10	NRC Review of Emergency Preparedness Plan	$0	$0	$0	$105	$26	$131
2.11	Prepare Post-Shutdown Decommissioning Activities Report (PSDAR)	$550	$1	$0	$0	$138	$688
2.12	NRC Review of Post-Shutdown Decommissioning Activities Report (PSDAR)	$0	$0	$0	$105	$26	$131
2.13	Respond to NRC quesitons on PSDAR	$34	$1	$0	$0	$9	$43
2.14	Prepare Decommissioning Cost Estimate (DCE)	$1,429	$4	$0	$0	$358	$1,791
2.15	NRC Review of Decommissioning Cost Estimate	$0	$0	$0	$105	$26	$131
2.16	Disposition of Legacy Wastes	$0	$0	$16,457	$0	$4,114	$20,571
2.17	Perform Historic Site Assessment and Site Characterization	$6,784	$838	$0	$0	$1,143	$8,765
2.18	Planning and Design For Cold and Dark	$9,716	$90	$0	$0	$2,451	$12,257
2.19	Implement Cold and Dark (Repower Site)	$16,141	$17,860	$0	$0	$8,500	$42,501
2.20	Install 12kV Service Line to Power Temporary Power Ring	$0	$0	$0	$5,250	$1,313	$6,563
2.21	Drain and De-Energize Non-Essential Systems (DEC Process)	$822	$183	$1,485	$0	$623	$3,114
2.22	Select Decommissioning General Contractor	$645	$8	$0	$0	$163	$817
2.23	Design Spent Fuel Pool Support System Modifications	$622	$8	$0	$0	$157	$787
2.24	Design Control Room Relocation	$601	$7	$0	$0	$152	$760
2.25	Design Spent Fuel Security System Modifications	$459	$5	$0	$0	$116	$580
2.26	Install Spent Fuel Pool System Modifications - Unit 2	$1,863	$4,101	$0	$0	$1,491	$7,456
2.27	Install Spent Fuel Pool System Modifications - Unit 3	$1,863	$4,101	$0	$0	$1,491	$7,456
2.28	Spent Fuel Pool System Modification Training	$0	$0	$0	$273	$68	$341
2.29	Implement Control Room Modifications	$1,004	$1,519	$0	$0	$631	$3,153
2.30	Implement Spent Fuel Pool Security Modifications	$525	$795	$0	$0	$330	$1,650
2.31	Transition Project Modifications	$0	$0	$0	$105	$26	$131

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2014 Dollars in Thousands

No	Item Description	Labor	Equipment	Disposal	Other	Contingency	Total
Distributed	**Subtotal**	**$48,154**	**$29,538**	**$17,942**	**$6,077**	**$24,665**	**$126,376**
Undistributed							
1.01	Utility Staff	$56,478	$0	$0	$0	$14,119	$70,597
1.02	Utility Staff HP Supplies	$0	$1,781	$0	$0	$445	$2,226
1.03	Security Guard Force	$2,087	$0	$0	$0	$522	$2,609
1.04	Security Related Expenses	$77	$0	$0	$0	$19	$96
1.05	Insurance	$0	$0	$0	$4,446	$1,111	$5,557
1.06	Site Lease and Easement Expenses	$0	$0	$0	$470	$70	$540
1.07	NRC Decommissioning Fees	$0	$0	$0	$2,390	$598	$2,988
1.08	Materials and Services	$0	$3,208	$0	$0	$802	$4,010
1.09	DAW Disposal	$0	$0	$295	$0	$74	$369
1.10	Energy	$0	$0	$0	$6,338	$1,584	$7,922
1.13	Craft Worker Training	$234	$0	$0	$0	$58	$292
1.14	Workers Compensation Insurance	$0	$0	$0	$283	$71	$353
1.15	Community Outreach	$1,638	$0	$0	$1,830	$867	$4,335
1.16	Property Tax	$0	$0	$0	$2,350	$588	$2,938
1.17	Association Fees and Expenses	$0	$2,350	$0	$0	$588	$2,938
1.18	Utilities (Water, gas, phone)	$0	$738	$0	$0	$185	$923
1.20	Non-Process Computers	$0	$157	$0	$0	$39	$196
1.21	Telecommunications	$0	$157	$0	$0	$39	$196
1.24	Environmental Permits and Fees	$0	$0	$0	$2,977	$744	$3,721
1.25	Decommissioning Advisor	$0	$0	$0	$1,567	$392	$1,958
Undistributed	**Subtotal**	**$60,513**	**$8,391**	**$295**	**$22,650**	**$22,915**	**$114,764**
Decon Pd 2	**Subtotal**	**$108,667**	**$37,928**	**$18,237**	**$28,727**	**$47,581**	**$241,140**

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Spent Fuel Alternative	Dry	Repository Opening Date:	1/1/2024		

2014 Dollars in Thousands

No	Item Description	Labor	Equipment	Disposal	Other	Contingency	Total
Decon Pd 3	**Decommissioning Preparations and Reactor Internals Segmentation**						
Distributed							
3.01	Prepare Integrated Work Sequence and Schedule for Decommissioning	$952	$0	$0	$0	$238	$1,190
3.02	Prepare Detailed Work Procedures and Activity Specifications for Decommissioning	$14,920	$70	$0	$0	$3,748	$18,738
3.03	Planning and Design of Primary System Decontamination	$516	$4	$0	$0	$130	$649
3.04	Planning and Design Site Infrastructure Improvements	$341	$4	$0	$0	$86	$431
3.05	Design Containment Access Modifications	$557	$6	$0	$0	$141	$705
3.06	Primary System Decontamination - Unit 2	$1,447	$1,857	$2,228	$0	$1,383	$6,914
3.07	Primary System Decontamination - Unit 3	$1,447	$1,857	$2,228	$0	$1,383	$6,914
3.08	Hot Spot Decontamination - Unit 2	$580	$887	$743	$0	$552	$2,761
3.09	Hot Spot Decontamination - Unit 3	$580	$913	$743	$0	$559	$2,794
3.10	Modify Containment Access- Unit 2	$315	$611	$0	$0	$231	$1,157
3.11	Modify Containment Access- Unit 3	$315	$611	$0	$0	$231	$1,157
3.12	Remove and Dispose of Missle Shields - Unit 2	$206	$30	$81	$0	$79	$395
3.13	Remove and Dispose of Reactor Head - Unit 2	$879	$453	$2,463	$0	$949	$4,744
3.14	Remove and Dispose of Missile Shields - Unit 3	$437	$178	$3,375	$0	$997	$4,987
3.15	Remove and Dispose of Reactor Head - Unit 3	$879	$453	$2,463	$0	$949	$4,744
3.16	Finalize Residual Radiation Inventory	$125	$0	$0	$287	$103	$516
3.17	Prepare Activity Specifications	$7,328	$696	$0	$0	$2,006	$10,031
3.18	Select Shipping Casks and Obtain Shipping Permits	$49	$0	$0	$0	$12	$62
3.19	Design, Specify, and Procure Special Items and Materials	$972	$5,379	$0	$0	$1,588	$7,938
3.22	Test Special Cutting and Handling Equipment and Train Operators	$1,157	$148	$0	$0	$326	$1,631
3.23	Finalize Internals and Vessel Segmenting Details - Unit 2	$212	$16	$0	$0	$57	$284
3.24	Segment, Package and Dispose of Reactor Internals - Unit 2	$5,669	$2,036	$62,661	$0	$17,591	$87,957
3.25	Transfer Internals Segmentation Equipment to Unit 3	$131	$19	$0	$0	$37	$187
3.26	Finalize Internals and Vessel Segmenting Details - Unit 3	$212	$16	$0	$0	$57	$284
3.27	Segment, Package and Dispose of Reactor Internals - Unit 3	$5,669	$2,036	$62,661	$0	$17,591	$87,957
3.28	Construct New Change Rooms, Hot Laundry, In-Plant Laydown Areas	$0	$1,290	$0	$0	$194	$1,484
3.29	Procure Non-Engineered Standard Equipment	$0	$5,454	$0	$0	$1,364	$6,818
Distributed	**Subtotal**	**$45,893**	**$25,024**	**$139,643**	**$287**	**$52,583**	**$263,431**
Undistributed							
1.01	Utility Staff	$79,350	$0	$0	$0	$19,837	$99,187

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2014 Dollars in Thousands

No	Item Description	Labor	Equipment	Disposal	Other	Contingency	Total
1.02	Utility Staff HP Supplies	$0	$2,693	$0	$0	$673	$3,366
1.03	Security Guard Force	$5,484	$0	$0	$0	$1,371	$6,855
1.04	Security Related Expenses	$326	$0	$0	$0	$82	$408
1.05	Insurance	$0	$0	$0	$8,000	$2,000	$10,000
1.06	Site Lease and Easement Expenses	$0	$0	$0	$1,235	$185	$1,420
1.07	NRC Decommissioning Fees	$0	$0	$0	$6,281	$1,570	$7,851
1.08	Materials and Services	$0	$4,582	$0	$0	$1,145	$5,727
1.09	DAW Disposal	$0	$0	$424	$0	$106	$529
1.10	Energy	$0	$0	$0	$10,226	$2,556	$12,782
1.11	Decommissioning General Contractor Staff	$62,219	$0	$0	$0	$15,555	$77,773
1.12	DGC HP Supplies	$0	$1,558	$0	$0	$389	$1,947
1.13	Craft Worker Training	$1,842	$0	$0	$0	$460	$2,302
1.14	Workers Compensation Insurance	$0	$0	$0	$742	$186	$928
1.15	Community Outreach	$4,303	$0	$0	$4,808	$2,278	$11,390
1.16	Property Tax	$0	$0	$0	$6,175	$1,544	$7,719
1.17	Association Fees and Expenses	$0	$6,175	$0	$0	$1,544	$7,719
1.18	Utilities (Water, gas, phone)	$0	$1,106	$0	$0	$277	$1,383
1.19	Tools and Equipment	$0	$182	$0	$0	$45	$227
1.20	Non-Process Computers	$0	$412	$0	$0	$103	$515
1.21	Telecommunications	$0	$412	$0	$0	$103	$515
1.22	Personal Computers	$0	$0	$0	$89	$22	$111
1.24	Environmental Permits and Fees	$0	$0	$0	$7,822	$1,955	$9,777
1.25	Decommissioning Advisor	$0	$0	$0	$4,117	$1,029	$5,146
Undistributed	**Subtotal**	**$153,524**	**$17,119**	**$424**	**$49,495**	**$55,017**	**$275,579**
Decon Pd 3	**Subtotal**	**$199,417**	**$42,144**	**$140,067**	**$49,782**	**$107,600**	**$539,009**

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2014 Dollars in Thousands

No	Item Description	Labor	Equipment	Disposal	Other	Contingency	Total
Decon Pd 4	**Plant Systems and Large Component Removal**						
Distributed							
4.01	Upgrade Rail Spur	$0	$0	$0	$3,277	$819	$4,096
4.02	Install GARDIAN System	$0	$0	$0	$525	$131	$656
4.03	Scaffolding for Non-Essential System Removal	$3,516	$1,144	$200	$0	$1,215	$6,075
4.04	Asbestos Abatement and Hazardous Waste Disposal for Non-Essential Systems - Unit	$0	$0	$0	$1,050	$525	$1,575
4.05	Lead Abatement for Non-Essential Systems Removal - Unit 2	$2,287	$23	$411	$0	$1,361	$4,082
4.06	Remove, Package and Dispose of Non-Essential Systems - Unit 2	$33,512	$5,597	$31,969	$0	$17,769	$88,847
4.07	Asbestos Abatement and Hazardous Waste Disposal for Non-Essential Systems - Unit	$0	$0	$0	$1,050	$525	$1,575
4.08	Lead Abatement for Non-Essential Systems - Unit 3	$2,287	$399	$411	$0	$1,549	$4,647
4.09	Remove, Package and Dispose of Non-Essential Systems - Unit 3	$36,851	$6,313	$36,610	$0	$19,944	$99,718
4.10	Remove Underground Diesel Tank - Unit 2	$111	$45	$0	$41	$49	$247
4.11	Remove Underground Diesel Tank - Unit 3	$111	$45	$0	$41	$49	$247
4.12	Remove and Dispose of Spent Fuel Storage Racks - Unit 2	$42	$36	$4,922	$0	$1,250	$6,250
4.13	Remove and Dispose of Spent Fuel Storage Racks - Unit 3	$42	$36	$4,922	$0	$1,250	$6,250
4.14	Remove and Dispose of Legacy Class B and C Waste - Unit 2	$0	$0	$500	$0	$125	$625
4.15	Remove and Dispose of Legacy Class B and C Waste - Unit 3	$0	$0	$500	$0	$125	$625
4.16	Drain Spent Fuel Pool and Process Liquid Waste - Unit 2	$557	$703	$0	$0	$315	$1,575
4.17	Drain Spent Fuel Pool and Process Liquid Waste - Unit 3	$557	$703	$0	$0	$315	$1,575
4.18	Segment, Package and Dispose of Spent Fuel Pool Island Equipment	$11	$2	$107	$0	$30	$150
4.19	Segment and Dispose of Fuel Pool Bridge Crane - Unit 2	$85	$12	$168	$0	$66	$332
4.20	Segment and Dispose of Fuel Pool Bridge Crane - Unit 3	$85	$12	$168	$0	$66	$332
4.21	Flush and Drain Essential Systems Following Fuel Pool Closure	$226	$181	$2,970	$0	$844	$4,221
4.22	Scaffolding for Essential System Removal	$989	$322	$56	$0	$342	$1,708
4.23	Asbestos Abatement and Hazardous Waste Disposal for Essential Systems	$0	$0	$0	$788	$394	$1,181
4.24	Lead Abatement for Essential Systems Removal	$332	$58	$59	$0	$225	$674
4.25	Remove, Package and Dispose of Essential Systems	$33,774	$5,869	$17,264	$0	$14,227	$71,134
4.26	Removal and Disposal of Spent Resins, Filter Media and Tank Sludge	$90	$40	$7,425	$0	$1,889	$9,445
4.27	Reactor Vessel Insulation Removal and Disposal - Unit 2	$105	$12	$147	$0	$66	$331
4.28	Segment, Package and Dispose of Reactor Pressure Vessel - Unit 2	$1,044	$2,834	$29,313	$0	$8,298	$41,489
4.29	Transfer Rx Vessel Segmentation Equipment to Unit 3	$122	$18	$0	$0	$35	$175
4.30	Procure Replacement Non-Engineered Standard Equipment	$0	$454	$0	$0	$114	$568
4.31	Reactor Vessel Insulation Removal and Disposal - Unit 3	$105	$12	$147	$0	$66	$331

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2014 Dollars in Thousands

No	Item Description	Labor	Equipment	Disposal	Other	Contingency	Total
4.32	Segment, Package and Dispose of Reactor Pressure Vessel - Unit 3	$1,044	$2,834	$29,313	$0	$8,298	$41,489
4.33	Remove and Dispose of Steam Generators - Unit 2	$2,789	$1,288	$18,154	$0	$5,558	$27,788
4.34	Remove and Dispose of Pressurizer - Unit 2	$462	$70	$2,620	$0	$788	$3,940
4.35	Remove and Dispose of Steam Generators - Unit 3	$2,789	$1,288	$18,154	$0	$5,558	$27,788
4.36	Remove and Dispose of Pressurizer - Unit 3	$462	$70	$2,620	$0	$788	$3,940
4.37	Remove and Dispose of Turbine Gantry Crane - Unit 2	$445	$229	$0	$4	$170	$848
4.38	Remove and Dispose of Turbine Gantry Crane - Unit 3	$445	$229	$0	$4	$170	$848
4.39	Prepare License Termination Plan	$1,646	$149	$0	$0	$449	$2,244
Distributed	**Subtotal**	**$126,926**	**$31,029**	**$209,131**	**$6,779**	**$95,755**	**$469,620**
Undistributed							
1.01	Utility Staff	$71,956	$0	$0	$0	$17,989	$89,945
1.02	Utility Staff HP Supplies	$0	$2,715	$0	$0	$679	$3,394
1.03	Security Guard Force	$4,638	$0	$0	$0	$1,159	$5,797
1.04	Security Related Expenses	$1,007	$0	$0	$0	$252	$1,259
1.05	Insurance	$0	$0	$0	$3,653	$913	$4,566
1.06	Site Lease and Easement Expenses	$0	$0	$0	$1,044	$157	$1,201
1.07	NRC Decommissioning Fees	$0	$0	$0	$5,312	$1,328	$6,639
1.08	Materials and Services	$0	$4,204	$0	$0	$1,051	$5,255
1.09	DAW Disposal	$0	$0	$1,568	$0	$392	$1,960
1.10	Energy	$0	$0	$0	$7,568	$1,892	$9,460
1.11	Decommissioning General Contractor Staff	$125,798	$0	$0	$0	$31,449	$157,247
1.12	DGC HP Supplies	$0	$5,834	$0	$0	$1,458	$7,292
1.13	Craft Worker Training	$7,788	$0	$0	$0	$1,947	$9,735
1.14	Workers Compensation Insurance	$0	$0	$0	$628	$157	$785
1.15	Community Outreach	$3,639	$0	$0	$4,066	$1,926	$9,632
1.16	Property Tax	$0	$0	$0	$5,222	$1,306	$6,528
1.18	Utilities (Water, gas, phone)	$0	$1,007	$0	$0	$252	$1,258
1.19	Tools and Equipment	$0	$423	$0	$0	$106	$529
1.20	Non-Process Computers	$0	$348	$0	$0	$87	$435
1.21	Telecommunications	$0	$348	$0	$0	$87	$435
1.24	Environmental Permits and Fees	$0	$0	$0	$6,615	$1,654	$8,268
1.25	Decommissioning Advisor	$0	$0	$0	$2,611	$653	$3,264

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2014 Dollars in Thousands

No	Item Description	Labor	Equipment	Disposal	Other	Contingency	Total
Undistributed	**Subtotal**	**$214,826**	**$14,879**	**$1,568**	**$36,718**	**$66,893**	**$334,884**
Decon Pd 4	**Subtotal**	**$341,752**	**$45,908**	**$210,699**	**$43,497**	**$162,649**	**$804,504**

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2014 Dollars in Thousands

No	Item Description	Labor	Equipment	Disposal	Other	Contingency	Total
Decon Pd 5	**Building Decontamination**						
Distributed							
5.01	Decon Containment Building - Unit 3	$6,056	$3,318	$54,825	$0	$16,050	$80,249
5.02	Decon Penetration Building - Unit 3	$1,065	$351	$2,933	$0	$1,087	$5,437
5.03	Decon Safety Equipment and MSIV Building - Unit 3	$905	$390	$5,562	$0	$1,715	$8,573
5.04	Decon Fuel Handling Building - Unit 3	$1,275	$577	$16,101	$0	$4,488	$22,442
5.05	Decon Turbine Building - Unit 3	$100	$95	$3,925	$0	$1,030	$5,150
5.06	Decon Containment Building - Unit 2	$6,056	$3,318	$54,825	$0	$16,050	$80,249
5.07	Decon Penetration Building - Unit 2	$1,065	$351	$2,933	$0	$1,087	$5,437
5.08	Decon Safety Equipment and MSIV Building - Unit 2	$911	$396	$5,777	$0	$1,771	$8,854
5.09	Decon Fuel Handling Building - Unit 2	$1,275	$577	$16,101	$0	$4,488	$22,442
5.10	Decon Turbine Building - Unit 2	$100	$95	$3,925	$0	$1,030	$5,150
5.11	Decon Auxiliary Radwaste Building - Common	$943	$691	$17,999	$0	$4,908	$24,541
5.12	Decon Auxiliary Control Building - Common	$198	$163	$38	$0	$100	$499
5.13	Decon Condensate Area and Tunnels - Units 2 & 3	$375	$316	$403	$0	$274	$1,368
5.14	Excavate, Remove and Dispose of Yard Area Drains	$1,159	$128	$240	$0	$382	$1,908
5.15	Remove and Dispose of Contaminated Sumps, Trenches and Pavement	$185	$21	$746	$0	$238	$1,191
5.16	Remove and Dispose of Radiologically Contaminated Soil	$192	$216	$1,158	$0	$392	$1,958
5.17	Segment, Package and Dispose of Contaminated Decon Equipment and Tooling	$38	$6	$92	$0	$34	$170
5.18	Radiological Survey of Structures During Decon	$4,702	$3,666	$0	$0	$1,255	$9,623
Distributed	**Subtotal**	**$26,600**	**$14,676**	**$187,585**	**$0**	**$56,379**	**$285,240**
Undistributed							
1.01	Utility Staff	$29,516	$0	$0	$0	$7,379	$36,895
1.02	Utility Staff HP Supplies	$0	$997	$0	$0	$249	$1,247
1.03	Security Guard Force	$2,520	$0	$0	$0	$630	$3,150
1.04	Security Related Expenses	$560	$0	$0	$0	$140	$701
1.05	Insurance	$0	$0	$0	$1,985	$496	$2,481
1.06	Site Lease and Easement Expenses	$0	$0	$0	$567	$85	$652
1.07	NRC Decommissioning Fees	$0	$0	$0	$2,886	$722	$3,608
1.08	Materials and Services	$0	$1,668	$0	$0	$417	$2,086
1.09	DAW Disposal	$0	$0	$464	$0	$116	$580
1.10	Energy	$0	$0	$0	$2,336	$584	$2,920

Table 1
SONGS Prompt DECON Base Case, 2024 DOE Acceptance, Dry Storage

		License Status	POL	Unit 2 Shut Down:	6/7/2013
Decommissioning Alternative	DECON	Fuel Pool Systems	Modified	Unit 3 Shut Down:	6/7/2013
Spent Fuel Alternative	Dry	Repository Opening Date:	1/1/2024		

2014 Dollars in Thousands

No	Item Description	Labor	Equipment	Disposal	Other	Contingency	Total
1.11	Decommissioning General Contractor Staff	$56,286	$0	$0	$0	$14,071	$70,357
1.12	DGC HP Supplies	$0	$3,170	$0	$0	$792	$3,962
1.13	Craft Worker Training	$1,693	$0	$0	$0	$423	$2,116
1.14	Workers Compensation Insurance	$0	$0	$0	$341	$85	$426
1.15	Community Outreach	$862	$0	$0	$964	$457	$2,283
1.16	Property Tax	$0	$0	$0	$2,837	$709	$3,547
1.18	Utilities (Water, gas, phone)	$0	$413	$0	$0	$103	$517
1.19	Tools and Equipment	$0	$204	$0	$0	$51	$255
1.20	Non-Process Computers	$0	$189	$0	$0	$47	$236
1.21	Telecommunications	$0	$189	$0	$0	$47	$236
1.22	Personal Computers	$0	$0	$0	$71	$18	$88
1.24	Environmental Permits and Fees	$0	$0	$0	$3,594	$899	$4,493
1.25	Decommissioning Advisor	$0	$0	$0	$825	$206	$1,031
Undistributed	**Subtotal**	**$91,437**	**$6,832**	**$464**	**$16,406**	**$28,728**	**$143,866**
Decon Pd 5	**Subtotal**	**$118,037**	**$21,508**	**$188,049**	**$16,406**	**$85,106**	**$429,106**

Table 1
SONGS Prompt DECON Base Case, 2024 DOE Acceptance, Dry Storage

		License Status	POL	Unit 2 Shut Down:	6/7/2013
Decommissioning Alternative	DECON	Fuel Pool Systems	Modified	Unit 3 Shut Down:	6/7/2013
Spent Fuel Alternative	Dry	Repository Opening Date:	1/1/2024		

2014 Dollars in Thousands

No	Item Description	Labor	Equipment	Disposal	Other	Contingency	Total
Decon Pd 6	**License Termination During Demolition**						
Distributed							
6.01	Final Status Survey	$9,613	$3,088	$0	$2,360	$2,259	$17,320
6.02	Prepare Final Report of Dismantling Program	$164	$4	$0	$0	$42	$210
Distributed	**Subtotal**	**$9,777**	**$3,091**	**$0**	**$2,360**	**$2,301**	**$17,530**
Undistributed							
1.01	Utility Staff	$1,378	$0	$0	$0	$345	$1,723
1.04	Security Related Expenses	$4	$0	$0	$0	$1	$5
1.07	NRC Decommissioning Fees	$0	$0	$0	$13,535	$3,384	$16,919
1.08	Materials and Services	$0	$47	$0	$0	$12	$58
1.09	DAW Disposal	$0	$0	$62	$0	$16	$78
1.10	Energy	$0	$0	$0	$1,872	$468	$2,340
1.11	Decommissioning General Contractor Staff	$651	$0	$0	$0	$163	$814
1.12	DGC HP Supplies	$0	$301	$0	$0	$75	$376
1.15	Community Outreach	$2,386	$0	$0	$2,666	$1,263	$6,315
1.18	Utilities (Water, gas, phone)	$0	$10	$0	$0	$3	$13
Undistributed	**Subtotal**	**$4,420**	**$357**	**$62**	**$18,074**	**$5,728**	**$28,641**
Decon Pd 6	**Subtotal**	**$14,197**	**$3,449**	**$62**	**$20,434**	**$8,029**	**$46,171**
A. License Termination	**Subtotal**	**$812,119**	**$150,936**	**$566,266**	**$171,959**	**$410,965**	**$2,112,246**

Table 1
SONGS Prompt DECON Base Case, 2024 DOE Acceptance, Dry Storage

		License Status	POL	Unit 2 Shut Down:	6/7/2013
Decommissioning Alternative	DECON	Fuel Pool Systems	Modified	Unit 3 Shut Down:	6/7/2013
Spent Fuel Alternative	Dry	Repository Opening Date:	1/1/2024		

2014 Dollars in Thousands

No	Item Description	Labor	Equipment	Disposal	Other	Contingency	Total
B. Spent Fuel							
SNF Pd 1	**Spent Fuel Management Transition**						
Distributed							
7.01	Security Shut Down Strategy	$0	$0	$0	$8,388	$0	$8,388
7.02	Design and Fabricate Spent Fuel Canisters	$0	$0	$0	$8,842	$0	$8,842
7.03	Post Fukushima Modifications - Unit 2	$0	$0	$0	$126	$0	$126
7.05	Cyber Security Modifications	$0	$0	$0	$1,901	$0	$1,901
Distributed	**Subtotal**	**$0**	**$0**	**$0**	**$19,258**	**$0**	**$19,258**
Undistributed							
2.01	Utility Spent Fuel Staff	$38,478	$0	$0	$0	$0	$38,478
2.04	Security Guard Force	$69,889	$0	$0	$0	$0	$69,889
2.09	Emergency Preparedness Fees	$0	$0	$0	$2,340	$0	$2,340
2.10	Spent Fuel Maintenance	$0	$0	$0	$32	$0	$32
Undistributed	**Subtotal**	**$108,367**	**$0**	**$0**	**$2,372**	**$0**	**$110,739**
SNF Pd 1	**Subtotal**	**$108,367**	**$0**	**$0**	**$21,630**	**$0**	**$129,997**

Table 1
SONGS Prompt DECON Base Case, 2024 DOE Acceptance, Dry Storage

		License Status	POL	Unit 2 Shut Down:	6/7/2013
Decommissioning Alternative	DECON	Fuel Pool Systems	Modified	Unit 3 Shut Down:	6/7/2013
Spent Fuel Alternative	Dry	Repository Opening Date:	1/1/2024		

2014 Dollars in Thousands

No	Item Description	Labor	Equipment	Disposal	Other	Contingency	Total
SNF Pd 2	**Spent Fuel Transfer to Dry Storage**						
Distributed							
8.01	Security Shut Down Strategy	$0	$0	$0	$2,855	$714	$3,569
8.02	Decay Heat Analysis	$0	$0	$0	$105	$26	$131
8.03	Zirconium Fire/ Shine Analysis	$0	$0	$0	$105	$26	$131
8.05	NRC Review of Irradiated Fuel Management Plan	$0	$0	$0	$105	$26	$131
8.07	ISFSI Pad Study	$0	$0	$0	$103	$26	$129
8.08	Design ISFSI Expansion	$0	$0	$0	$3,150	$788	$3,938
8.09	Construct ISFSI Expansion	$0	$0	$0	$33,600	$8,400	$42,000
8.10	Purchase and Fabrication of Spent Fuel Canisters and AHSMs - Unit 2	$0	$49,613	$0	$0	$12,403	$62,016
8.11	Purchase and Fabrication Spent Fuel Canisters and AHSMs - Unit 3	$0	$50,794	$0	$0	$12,698	$63,492
8.12	Deliver and Load Spent Fuel Canisters and Transfer to ISFSI - Unit 2	$71,338	$17,478	$0	$0	$22,204	$111,021
8.13	Deliver and Load Spent Fuel Canisters and Transfer to ISFSI - Unit 3	$73,037	$17,894	$0	$0	$22,733	$113,664
Distributed	**Subtotal**	**$144,375**	**$135,779**	**$0**	**$40,023**	**$80,044**	**$400,221**
Undistributed							
2.01	Utility Spent Fuel Staff	$90,824	$0	$0	$0	$22,706	$113,530
2.02	Utility Staff HP Supplies	$0	$6,590	$0	$0	$1,647	$8,237
2.04	Security Guard Force	$112,313	$0	$0	$0	$28,078	$140,391
2.05	Security Related Expenses	$1,334	$0	$0	$0	$333	$1,667
2.06	Insurance	$0	$0	$0	$4,408	$1,102	$5,510
2.08	NRC Spent Fuel Fees	$0	$0	$0	$1,107	$277	$1,383
2.09	Emergency Preparedness Fees	$0	$0	$0	$18,756	$4,689	$23,445
2.10	Spent Fuel Maintenance	$0	$0	$0	$2,131	$533	$2,664
2.11	Materials and Services	$0	$5,848	$0	$0	$1,462	$7,310
2.12	DAW Disposal	$0	$0	$275	$0	$69	$343
2.13	Energy	$0	$0	$0	$3,991	$998	$4,989
2.15	Craft Worker Training	$2,119	$0	$0	$0	$530	$2,649
2.18	Utilities (Water, gas, phone)	$0	$3,572	$0	$0	$893	$4,465
2.22	Personal Computers	$0	$0	$0	$14	$3	$17
Undistributed	**Subtotal**	**$206,590**	**$16,010**	**$275**	**$30,406**	**$63,320**	**$316,601**
SNF Pd 2	**Subtotal**	**$350,965**	**$151,789**	**$275**	**$70,429**	**$143,364**	**$716,822**

Table 1
SONGS Prompt DECON Base Case, 2024 DOE Acceptance, Dry Storage

		License Status	POL	Unit 2 Shut Down:	6/7/2013
Decommissioning Alternative	DECON	Fuel Pool Systems	Modified	Unit 3 Shut Down:	6/7/2013
Spent Fuel Alternative	Dry	Repository Opening Date:	1/1/2024		

2014 Dollars in Thousands

No	Item Description	Labor	Equipment	Disposal	Other	Contingency	Total
SNF Pd 3	**Dry Storage During Decommissioning - Units 1, 2 and 3**						
Undistributed							
2.01	Utility Spent Fuel Staff	$39,894	$0	$0	$0	$9,973	$49,867
2.02	Utility Staff HP Supplies	$0	$1,487	$0	$0	$372	$1,859
2.04	Security Guard Force	$45,944	$0	$0	$0	$11,486	$57,430
2.05	Security Related Expenses	$2,556	$0	$0	$0	$639	$3,195
2.08	NRC Spent Fuel Fees	$0	$0	$0	$2,302	$576	$2,878
2.10	Spent Fuel Maintenance	$0	$0	$0	$1,478	$370	$1,848
2.11	Materials and Services	$0	$2,017	$0	$0	$504	$2,522
2.13	Energy	$0	$0	$0	$1,209	$302	$1,511
2.18	Utilities (Water, gas, phone)	$0	$1,380	$0	$0	$345	$1,725
2.22	Personal Computers	$0	$0	$0	$12	$3	$15
Undistributed	**Subtotal**	**$88,393**	**$4,884**	**$0**	**$5,001**	**$24,570**	**$122,849**
SNF Pd 3	**Subtotal**	**$88,393**	**$4,884**	**$0**	**$5,001**	**$24,570**	**$122,849**

Table 1
SONGS Prompt DECON Base Case, 2024 DOE Acceptance, Dry Storage

		License Status	POL	Unit 2 Shut Down:	6/7/2013
Decommissioning Alternative	DECON	Fuel Pool Systems	Modified	Unit 3 Shut Down:	6/7/2013
Spent Fuel Alternative	Dry	Repository Opening Date:	1/1/2024		

2014 Dollars in Thousands

No	Item Description	Labor	Equipment	Disposal	Other	Contingency	Total
SNF Pd 4	**Dry Storage Only - Units 1, 2 and 3**						
Undistributed							
2.01	Utility Spent Fuel Staff	$12,687	$0	$0	$0	$3,172	$15,859
2.02	Utility Staff HP Supplies	$0	$882	$0	$0	$220	$1,102
2.03	Additional Staff for Spent Fuel Shipping	$1,119	$0	$0	$0	$280	$1,398
2.04	Security Guard Force	$14,949	$0	$0	$0	$3,737	$18,687
2.05	Security Related Expenses	$2,506	$0	$0	$0	$626	$3,132
2.06	Insurance	$0	$0	$0	$2,538	$634	$3,172
2.07	Site Lease and Easement Expenses	$0	$0	$0	$1,154	$173	$1,327
2.08	NRC Spent Fuel Fees	$0	$0	$0	$1,638	$409	$2,047
2.10	Spent Fuel Maintenance	$0	$0	$0	$481	$120	$601
2.11	Materials and Services	$0	$778	$0	$0	$194	$972
2.13	Energy	$0	$0	$0	$393	$98	$492
2.16	Workers Compensation Insurance	$0	$0	$0	$694	$173	$867
2.17	Property Tax	$0	$0	$0	$6,412	$1,603	$8,015
2.18	Utilities (Water, gas, phone)	$0	$475	$0	$0	$119	$594
2.20	Non-Process Computers	$0	$192	$0	$0	$48	$240
2.21	Telecommunications	$0	$192	$0	$0	$48	$240
2.22	Personal Computers	$0	$0	$0	$15	$4	$18
Undistributed	**Subtotal**	**$31,261**	**$2,519**	**$0**	**$13,325**	**$11,661**	**$58,765**
SNF Pd 4	**Subtotal**	**$31,261**	**$2,519**	**$0**	**$13,325**	**$11,661**	**$58,765**

Table 1
SONGS Prompt DECON Base Case, 2024 DOE Acceptance, Dry Storage

		License Status	POL	Unit 2 Shut Down:	6/7/2013
Decommissioning Alternative	DECON	Fuel Pool Systems	Modified	Unit 3 Shut Down:	6/7/2013
Spent Fuel Alternative	Dry	Repository Opening Date:	1/1/2024		

2014 Dollars in Thousands

No	Item Description	Labor	Equipment	Disposal	Other	Contingency	Total
SNF Pd 5	**Dry Storage Only - Units 2 and 3**						
Undistributed							
2.01	Utility Spent Fuel Staff	$48,480	$0	$0	$0	$12,120	$60,601
2.02	Utility Staff HP Supplies	$0	$3,369	$0	$0	$842	$4,211
2.03	Additional Staff for Spent Fuel Shipping	$4,275	$0	$0	$0	$1,069	$5,344
2.04	Security Guard Force	$57,126	$0	$0	$0	$14,281	$71,407
2.05	Security Related Expenses	$4,124	$0	$0	$0	$1,031	$5,155
2.06	Insurance	$0	$0	$0	$9,698	$2,425	$12,123
2.07	Site Lease and Easement Expenses	$0	$0	$0	$4,409	$661	$5,071
2.08	NRC Spent Fuel Fees	$0	$0	$0	$6,259	$1,565	$7,823
2.10	Spent Fuel Maintenance	$0	$0	$0	$1,838	$459	$2,297
2.11	Materials and Services	$0	$2,972	$0	$0	$743	$3,715
2.13	Energy	$0	$0	$0	$1,503	$376	$1,879
2.16	Workers Compensation Insurance	$0	$0	$0	$2,651	$663	$3,314
2.17	Property Tax	$0	$0	$0	$22,053	$5,513	$27,566
2.18	Utilities (Water, gas, phone)	$0	$1,816	$0	$0	$454	$2,270
2.20	Non-Process Computers	$0	$735	$0	$0	$184	$919
2.21	Telecommunications	$0	$735	$0	$0	$184	$919
2.22	Personal Computers	$0	$0	$0	$32	$8	$40
Undistributed	**Subtotal**	**$114,005**	**$9,627**	**$0**	**$48,443**	**$42,578**	**$214,653**
SNF Pd 5	**Subtotal**	**$114,005**	**$9,627**	**$0**	**$48,443**	**$42,578**	**$214,653**

Table 1
SONGS Prompt DECON Base Case, 2024 DOE Acceptance, Dry Storage

		License Status	POL	Unit 2 Shut Down:	6/7/2013
Decommissioning Alternative	DECON	Fuel Pool Systems	Modified	Unit 3 Shut Down:	6/7/2013
Spent Fuel Alternative	Dry	Repository Opening Date:	1/1/2024		

2014 Dollars in Thousands

No	Item Description	Labor	Equipment	Disposal	Other	Contingency	Total
SNF D&D Pd 1	**ISFSI License Termination**						
Distributed							
12.01	Preparation and NRC Review of License Termination Plan	$116	$0	$0	$163	$70	$349
Distributed	**Subtotal**	**$116**	**$0**	**$0**	**$163**	**$70**	**$349**
Undistributed							
2.01	Utility Spent Fuel Staff	$366	$0	$0	$0	$91	$457
2.02	Utility Staff HP Supplies	$0	$11	$0	$0	$3	$14
2.04	Security Guard Force	$181	$0	$0	$0	$45	$226
2.05	Security Related Expenses	$70	$0	$0	$0	$18	$88
2.06	Insurance	$0	$0	$0	$215	$54	$269
2.07	Site Lease and Easement Expenses	$0	$0	$0	$98	$15	$112
2.08	NRC Spent Fuel Fees	$0	$0	$0	$75	$19	$94
2.11	Materials and Services	$0	$17	$0	$0	$4	$21
2.13	Energy	$0	$0	$0	$102	$26	$128
2.16	Workers Compensation Insurance	$0	$0	$0	$59	$15	$73
2.17	Property Tax	$0	$0	$0	$543	$136	$679
2.18	Utilities (Water, gas, phone)	$0	$7	$0	$0	$2	$9
Undistributed	**Subtotal**	**$617**	**$36**	**$0**	**$1,092**	**$426**	**$2,172**
SNF D&D Pd 1	**Subtotal**	**$733**	**$36**	**$0**	**$1,255**	**$496**	**$2,520**

Table 1
SONGS Prompt DECON Base Case, 2024 DOE Acceptance, Dry Storage

		License Status	POL	Unit 2 Shut Down:	6/7/2013
Decommissioning Alternative	DECON	Fuel Pool Systems	Modified	Unit 3 Shut Down:	6/7/2013
Spent Fuel Alternative	Dry	Repository Opening Date:	1/1/2024		

2014 Dollars in Thousands

No	Item Description	Labor	Equipment	Disposal	Other	Contingency	Total
SNF D&D Pd 2	**ISFSI Demolition**						
Distributed							
13.01	Install GARDIAN Bulk Assay System	$0	$0	$0	$525	$131	$656
13.02	Decon AHSMs	$339	$147	$443	$0	$232	$1,161
13.03	Final Status Survey of ISFSI	$1,589	$256	$0	$0	$277	$2,122
13.04	Clean Demolition of ISFSI AHSMs and Pad	$4,094	$2,590	$3,333	$0	$2,504	$12,521
13.05	Clean Demolition of ISFSI Support Structures	$1,126	$458	$1,372	$0	$739	$3,696
13.06	Restore ISFSI Site	$246	$161	$0	$0	$102	$509
13.07	Preparation of Final Report on Decommissioning and NRC Review	$52	$0	$0	$0	$13	$65
Distributed	**Subtotal**	**$7,446**	**$3,612**	**$5,148**	**$525**	**$3,998**	**$20,729**
Undistributed							
2.01	Utility Spent Fuel Staff	$1,801	$0	$0	$0	$450	$2,251
2.02	Utility Staff HP Supplies	$0	$72	$0	$0	$18	$90
2.04	Security Guard Force	$704	$0	$0	$0	$176	$880
2.05	Security Related Expenses	$37	$0	$0	$0	$9	$46
2.11	Materials and Services	$0	$93	$0	$0	$23	$116
2.12	DAW Disposal	$0	$0	$7	$0	$2	$8
2.13	Energy	$0	$0	$0	$268	$67	$334
2.14	Decommissioning General Contractor Staff	$4,525	$0	$0	$0	$1,131	$5,656
2.15	Craft Worker Training	$189	$0	$0	$0	$47	$236
2.18	Utilities (Water, gas, phone)	$0	$35	$0	$0	$9	$43
2.24	DGC HP Supplies	$0	$159	$0	$0	$40	$199
Undistributed	**Subtotal**	**$7,255**	**$359**	**$7**	**$268**	**$1,972**	**$9,861**
SNF D&D Pd 2	**Subtotal**	**$14,701**	**$3,972**	**$5,154**	**$793**	**$5,970**	**$30,590**
B. Spent Fuel	**Subtotal**	**$708,425**	**$172,826**	**$5,429**	**$160,876**	**$228,639**	**$1,276,196**

Table 1
SONGS Prompt DECON Base Case, 2024 DOE Acceptance, Dry Storage

		License Status	POL	Unit 2 Shut Down:	6/7/2013
Decommissioning Alternative	DECON	Fuel Pool Systems	Modified	Unit 3 Shut Down:	6/7/2013
Spent Fuel Alternative	Dry	Repository Opening Date:	1/1/2024		

2014 Dollars in Thousands

No	Item Description	Labor	Equipment	Disposal	Other	Contingency	Total
C. Site Restoration							
SR Pd 1	**Transition to Site Restoration**						
Distributed							
14.01	Mesa Site Phase I and II Site Assessment	$0	$0	$0	$42	$11	$53
14.02	Disposition Hazardous Waste from Mesa Site	$0	$0	$0	$211	$106	$317
14.03	Mesa Site Characterization Survey	$988	$261	$0	$0	$312	$1,561
14.04	Fuel Cancellation Expense	$0	$0	$0	$17,679	$0	$17,679
Distributed	**Subtotal**	**$988**	**$261**	**$0**	**$17,932**	**$428**	**$19,610**
Undistributed							
3.05	Site Lease and Easement Expenses	$0	$0	$0	$1,030	$0	$1,030
3.11	Severance	$0	$0	$0	$109,850	$0	$109,850
Undistributed	**Subtotal**	**$0**	**$0**	**$0**	**$110,880**	**$0**	**$110,880**
SR Pd 1	**Subtotal**	**$988**	**$261**	**$0**	**$128,812**	**$428**	**$130,489**

Table 1
SONGS Prompt DECON Base Case, 2024 DOE Acceptance, Dry Storage

		License Status	POL	Unit 2 Shut Down:	6/7/2013
Decommissioning Alternative	DECON	Fuel Pool Systems	Modified	Unit 3 Shut Down:	6/7/2013
Spent Fuel Alternative	Dry	Repository Opening Date:	1/1/2024		

2014 Dollars in Thousands

No	Item Description	Labor	Equipment	Disposal	Other	Contingency	Total
SR Pd 2	**Building Demolition During Decommissioning**						
Distributed							
15.01	Prepare Site Restoration Demolition Plan and Schedule	$684	$10	$0	$0	$173	$866
15.02	Obtain Required Permits For Mesa, South Access and South Yard	$209	$4	$0	$0	$53	$266
15.03	Demolish Service Building (K-10, 20, 30)	$250	$189	$481	$0	$230	$1,150
15.04	Demolish South Security Processing Facility (K-70)	$46	$44	$122	$0	$53	$264
15.05	Demolish Staging Warehouse	$67	$55	$126	$0	$62	$311
15.06	Demolish Administration Building (K-40/50)	$367	$258	$565	$0	$297	$1,487
15.07	Demolish South Yard Area Buildings T-10, 20, 60 and Haz Mat.	$670	$590	$1,370	$0	$658	$3,288
15.08	Demolish REMS Staging Pad	$98	$184	$549	$0	$208	$1,038
15.09	Demolish Mesa Buildings	$2,788	$1,879	$6,006	$0	$2,668	$13,341
15.10	Remove Underground Fuel Storage Tanks	$56	$22	$0	$21	$25	$123
15.11	Demolish Mesa Roads and Parking Lots	$582	$400	$0	$0	$245	$1,227
15.12	Finish Grading and Re-vegetate Mesa Site	$299	$404	$0	$0	$176	$878
Distributed	**Subtotal**	**$6,114**	**$4,038**	**$9,219**	**$21**	**$4,848**	**$24,239**
Undistributed							
3.01	Utility Staff	$2,563	$0	$0	$0	$641	$3,204
3.03	Security Related Expenses	$898	$0	$0	$0	$224	$1,122
3.05	Site Lease and Easement Expenses	$0	$0	$0	$4,266	$640	$4,906
3.06	Materials and Services	$0	$134	$0	$0	$34	$168
3.08	Decommissioning General Contractor Staff	$4,248	$0	$0	$0	$1,062	$5,310
3.09	Craft Worker Training	$318	$0	$0	$0	$80	$398
3.11	Severance	$0	$0	$0	$8,688	$2,172	$10,860
3.13	Utilities (Water, gas, phone)	$0	$29	$0	$0	$7	$36
Undistributed	**Subtotal**	**$8,027**	**$164**	**$0**	**$12,955**	**$4,860**	**$26,005**
SR Pd 2	**Subtotal**	**$14,141**	**$4,201**	**$9,219**	**$12,975**	**$9,708**	**$50,245**

Table 1
SONGS Prompt DECON Base Case, 2024 DOE Acceptance, Dry Storage

		License Status	POL	Unit 2 Shut Down:	6/7/2013
Decommissioning Alternative	DECON	Fuel Pool Systems	Modified	Unit 3 Shut Down:	6/7/2013
Spent Fuel Alternative	Dry	Repository Opening Date:	1/1/2024		

2014 Dollars in Thousands

No	Item Description	Labor	Equipment	Disposal	Other	Contingency	Total
SR Pd 3	**Subsurface Demolition Engineering and Permitting**						
Distributed							
16.01	Hydrogeologic Investigation and Outfall Conduit Survey	$297	$131	$0	$105	$133	$667
16.02	Subsurface Structure Removal Engineering Planning and Design	$1,264	$33	$0	$0	$324	$1,621
16.03	Environmental Impacts Analyses for Lease Termination Activities	$581	$50	$0	$525	$289	$1,445
16.04	Final Site Grading and Shoreline Protection Engineering Planning and Design	$242	$13	$0	$0	$64	$319
16.05	Obtain Required Permits and Approvals	$1,856	$20	$0	$263	$535	$2,673
Distributed	**Subtotal**	**$4,240**	**$248**	**$0**	**$893**	**$1,345**	**$6,726**
Undistributed							
3.03	Security Related Expenses	$275	$0	$0	$0	$69	$344
3.11	Severance	$0	$0	$0	$24,674	$6,168	$30,842
Undistributed	**Subtotal**	**$275**	**$0**	**$0**	**$24,674**	**$6,237**	**$31,186**
SR Pd 3	**Subtotal**	**$4,516**	**$248**	**$0**	**$25,566**	**$7,582**	**$37,912**

Table 1
SONGS Prompt DECON Base Case, 2024 DOE Acceptance, Dry Storage

		License Status	POL	Unit 2 Shut Down:	6/7/2013
Decommissioning Alternative	DECON	Fuel Pool Systems	Modified	Unit 3 Shut Down:	6/7/2013
Spent Fuel Alternative	Dry	Repository Opening Date:	1/1/2024		

2014 Dollars in Thousands

No	Item Description	Labor	Equipment	Disposal	Other	Contingency	Total
SR Pd 4	**Building Demolition to 3 Feet Below Grade**						
Distributed							
17.01	Procure Clean Building Demolition Equipment	$0	$10,691	$0	$0	$2,673	$13,363
17.02	Install Temporary Structures	$11	$190	$0	$0	$30	$230
17.03	Install Erosion and Sediment Controls	$123	$14	$0	$0	$34	$172
17.04	Remove Cathodic Protection Trench	$1,813	$1,527	$22	$0	$840	$4,201
17.05	Remove Protected Area Security Fencing	$57	$18	$0	$0	$19	$95
17.06	Remove Protected Area Pavement	$139	$97	$755	$0	$248	$1,239
17.07	Detension and Remove Unit 3 Containment Builidng Tendons	$0	$0	$0	$4,200	$1,050	$5,250
17.08	Demolish Diesel Generator Building - Unit 3	$618	$245	$794	$0	$414	$2,072
17.09	Demolish Condensate Building and Transformer Pads - Unit 3	$1,067	$1,755	$3,183	$0	$1,501	$7,505
17.10	Demolish Full Flow Area and Turbine Building - Unit 3	$3,221	$1,149	$3,444	$0	$1,953	$9,767
17.11	Demolish Unit 3 Fuel Handling Building to 3-Feet Below Grade	$306	$354	$1,470	$0	$533	$2,663
17.12	Demolish Penetration Building - Unit 3	$293	$167	$642	$0	$275	$1,377
17.13	Demolish Safety Equipment and MSIV Building - Unit 3	$336	$403	$1,858	$0	$649	$3,246
17.14	Demolish Unit 3 Containment Building to 3-Feet Below Grade	$2,418	$1,351	$6,198	$0	$2,492	$12,459
17.15	Detension and Remove Unit 2 Containment Builidng Tendons	$0	$0	$0	$4,200	$1,050	$5,250
17.16	Demolish Diesel Generator Building - Unit 2	$128	$168	$787	$0	$271	$1,353
17.17	Demolish Condensate Building and Transformer Pads - Unit 2	$1,067	$1,755	$3,183	$0	$1,501	$7,505
17.18	Demolish Full Flow Area and Turbine Building - Unit 2	$3,734	$1,186	$3,447	$0	$2,092	$10,458
17.19	Demolish Unit 2 Fuel Handling Building to 3-Feet Below Grade	$306	$354	$1,470	$0	$533	$2,663
17.20	Demolish Penetration Building - Unit 2	$99	$136	$639	$0	$219	$1,093
17.21	Demolish Safety and MSIV Equipment Building - Unit 2	$336	$403	$1,859	$0	$649	$3,247
17.22	Demolish Unit 2 Containment Building to 3-Feet Below Grade	$2,418	$1,351	$6,198	$0	$2,492	$12,459
17.23	Demolish AWS Building	$1,108	$1,050	$2,925	$0	$1,271	$6,354
17.24	Demolish Building L-50	$59	$33	$67	$0	$40	$198
17.25	Demolish Maintenance Building 4 (B-64/B-65)	$24	$13	$25	$0	$16	$78
17.26	Demolish Maintenance Building 5 (B-62/B-63)	$35	$20	$37	$0	$23	$115
17.27	Demolish Outage Control Center	$98	$57	$148	$0	$76	$378
17.28	Demolish Maintenance Building 2 (B-49/B-50)	$49	$32	$82	$0	$41	$205
17.29	Demolish Maintenance Building 1 (B-43/B-44)	$163	$196	$857	$0	$304	$1,520
17.30	Demolish Auxilary Radwaste Building - Common	$1,521	$1,984	$9,214	$0	$3,180	$15,898
17.31	Demolish Auxilary Control Building - Common	$1,491	$811	$3,219	$0	$1,380	$6,901

Table 1
SONGS Prompt DECON Base Case, 2024 DOE Acceptance, Dry Storage

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Decommissioning Alternative	DECON	Fuel Pool Systems	Modified	Unit 3 Shut Down:	6/7/2013
Spent Fuel Alternative	Dry	Repository Opening Date:	1/1/2024		

2014 Dollars in Thousands

No	Item Description	Labor	Equipment	Disposal	Other	Contingency	Total
17.32	Remove Systems and Demolish Make-Up Demineralizer Structures	$737	$122	$471	$0	$332	$1,662
17.33	Install Concrete Plugs in Intake and Discharge Structures	$272	$1,614	$0	$0	$472	$2,358
17.34	Demolish Intake and Discharge Structures to 3-Feet Below Grade	$82	$114	$535	$0	$183	$914
Distributed	**Subtotal**	**$24,128**	**$29,358**	**$53,530**	**$8,400**	**$28,834**	**$144,249**
Undistributed							
3.01	Utility Staff	$12,553	$0	$0	$0	$3,138	$15,691
3.02	Security Guard Force	$2,480	$0	$0	$0	$620	$3,100
3.03	Security Related Expenses	$1,158	$0	$0	$0	$290	$1,448
3.04	Insurance	$0	$0	$0	$3,995	$999	$4,993
3.05	Site Lease and Easement Expenses	$0	$0	$0	$1,340	$201	$1,541
3.06	Materials and Services	$0	$751	$0	$0	$188	$938
3.07	Energy	$0	$0	$0	$1,184	$296	$1,480
3.08	Decommissioning General Contractor Staff	$50,906	$0	$0	$0	$12,727	$63,633
3.09	Craft Worker Training	$1,999	$0	$0	$0	$500	$2,498
3.10	Workers Compensation Insurance	$0	$0	$0	$806	$201	$1,007
3.11	Severance	$0	$0	$0	$7,273	$1,818	$9,091
3.12	Property Tax	$0	$0	$0	$6,701	$1,675	$8,377
3.13	Utilities (Water, gas, phone)	$0	$214	$0	$0	$53	$267
3.14	Tools and Equipment	$0	$156	$0	$0	$39	$195
3.15	Non-Process Computers	$0	$223	$0	$0	$56	$279
3.16	Telecommunications	$0	$223	$0	$0	$56	$279
Undistributed	**Subtotal**	**$69,096**	**$1,567**	**$0**	**$21,298**	**$22,856**	**$114,817**
SR Pd 4	**Subtotal**	**$93,224**	**$30,924**	**$53,530**	**$29,698**	**$51,690**	**$259,066**

Table 1
SONGS Prompt DECON Base Case, 2024 DOE Acceptance, Dry Storage

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Spent Fuel Alternative	Dry	Repository Opening Date:	1/1/2024		

2014 Dollars in Thousands

No	Item Description	Labor	Equipment	Disposal	Other	Contingency	Total
SR Pd 5	**Subgrade Structure Removal Below - 3 Feet**						
Distributed							
18.01	Procure Subsurface Structure Demolition Equipment	$0	$6,630	$0	$0	$1,658	$8,288
18.02	Install Sheet Piling and Excavation Shoring	$8,468	$17,219	$0	$0	$6,422	$32,109
18.03	Install Dewatering System and Effluent Treatment and Discharge Controls	$0	$0	$0	$9,651	$2,413	$12,064
18.04	Demolish and Backfill Unit 3 Condensate Storage Area Below -3 Feet	$179	$305	$912	$0	$349	$1,746
18.05	Demolish and Backfill Unit 3 Diesel Generator Builidng Below -3 Feet	$130	$173	$442	$0	$186	$932
18.06	Demolish and Backfill Unit 3 Fuel Handling Building Below -3 Feet	$271	$696	$1,170	$0	$534	$2,671
18.07	Demolish and Backfill Unit 3 Radwaste and Control Building Below -3 Feet	$1,367	$3,268	$5,249	$0	$2,471	$12,355
18.08	Demolish and Backfill Unit 3 Turbine Building Structure Below 9 Ft Elevation	$3,956	$9,277	$12,551	$0	$6,446	$32,231
18.09	Demolish and Backfill Unit 3 Safety Equipment Building Below -3 Feet	$717	$1,883	$2,713	$0	$1,328	$6,641
18.10	Demolish and Backfill Unit 3 Penetration Area Below -3 Feet	$294	$586	$1,285	$0	$541	$2,706
18.11	Demolish and Backfill Unit 3 Full Flow Building Below -3 Feet	$167	$527	$411	$0	$276	$1,382
18.12	Demolish and Backfill Unit 3 Containment Building Below -3 Feet	$1,211	$2,214	$4,636	$0	$2,015	$10,077
18.13	Demolish and Backfill Unit 2 Condensate Storage Area Below -3 Feet	$179	$305	$912	$0	$349	$1,746
18.14	Demolish and Backfill Unit 2 Diesel Generator Builidng Below -3 Feet	$130	$173	$442	$0	$186	$932
18.15	Demolish and Backfill Unit 2 Fuel Handling Building Below -3 Feet	$271	$696	$1,170	$0	$534	$2,671
18.16	Demolish and Backfill Unit 2 Radwaste and Control Building Below -3 Feet	$1,415	$3,308	$5,249	$0	$2,493	$12,466
18.17	Demolish and Backfill Unit 2 Turbine Building Structure Below 9 Ft Elevation	$3,959	$9,277	$12,551	$0	$6,447	$32,234
18.18	Demolish and Backfill Unit 2 Safety Equipment Building Below -3 Feet	$717	$1,883	$2,713	$0	$1,328	$6,641
18.19	Demolish and Backfill Unit 2 Penetration Area Below -3 Feet	$294	$586	$1,285	$0	$541	$2,706
18.20	Demolish and Backfill Unit 2 Full Flow Building Below -3 Feet	$167	$527	$411	$0	$276	$1,382
18.21	Demolish and Backfill Unit 2 Containment Building Below -3 Feet	$1,211	$2,214	$4,636	$0	$2,015	$10,077
18.22	Demolish and Backfill Intake Structure Below -3 Feet	$6,664	$12,970	$36,706	$0	$14,085	$70,426
18.23	Remove Off Shore Intake and Outfall Conduits	$12,406	$44,308	$19,580	$0	$19,073	$95,367
18.24	Remove Sheet Piling and Excavation Shoring	$11,776	$0	$0	$0	$2,944	$14,721
18.25	Remove Dewatering System and Effluent Treatment	$0	$0	$0	$2,308	$577	$2,885
18.26	Finish Grading and Re-Vegetate Site	$945	$813	$0	$0	$440	$2,198
18.27	Remove Temporary Structures	$58	$48	$0	$0	$16	$122
Distributed	**Subtotal**	**$56,952**	**$119,889**	**$115,025**	**$11,959**	**$75,946**	**$379,772**
Undistributed							
3.01	Utility Staff	$7,082	$0	$0	$0	$1,771	$8,853

Table 1
SONGS Prompt DECON Base Case, 2024 DOE Acceptance, Dry Storage

		License Status	POL	Unit 2 Shut Down:	6/7/2013
Decommissioning Alternative	DECON	Fuel Pool Systems	Modified	Unit 3 Shut Down:	6/7/2013
Spent Fuel Alternative	Dry	Repository Opening Date:	1/1/2024		

2014 Dollars in Thousands

No	Item Description	Labor	Equipment	Disposal	Other	Contingency	Total
3.02	Security Guard Force	$1,830	$0	$0	$0	$458	$2,288
3.03	Security Related Expenses	$139	$0	$0	$0	$35	$173
3.04	Insurance	$0	$0	$0	$2,948	$737	$3,685
3.05	Site Lease and Easement Expenses	$0	$0	$0	$989	$148	$1,137
3.06	Materials and Services	$0	$415	$0	$0	$104	$519
3.07	Energy	$0	$0	$0	$814	$204	$1,018
3.08	Decommissioning General Contractor Staff	$26,176	$0	$0	$0	$6,544	$32,720
3.09	Craft Worker Training	$983	$0	$0	$0	$246	$1,229
3.10	Workers Compensation Insurance	$0	$0	$0	$595	$149	$743
3.11	Severance	$0	$0	$0	$2,050	$513	$2,563
3.12	Property Tax	$0	$0	$0	$4,946	$1,237	$6,183
3.13	Utilities (Water, gas, phone)	$0	$128	$0	$0	$32	$160
3.14	Tools and Equipment	$0	$73	$0	$0	$18	$91
3.15	Non-Process Computers	$0	$165	$0	$0	$41	$206
3.16	Telecommunications	$0	$165	$0	$0	$41	$206
Undistributed	**Subtotal**	**$36,211**	**$946**	**$0**	**$12,343**	**$12,276**	**$61,775**
SR Pd 5	**Subtotal**	**$93,163**	**$120,834**	**$115,025**	**$24,302**	**$88,222**	**$441,547**

Table 1
SONGS Prompt DECON Base Case, 2024 DOE Acceptance, Dry Storage

		License Status	POL	Unit 2 Shut Down:	6/7/2013
Decommissioning Alternative	DECON	Fuel Pool Systems	Modified	Unit 3 Shut Down:	6/7/2013
Spent Fuel Alternative	Dry	Repository Opening Date:	1/1/2024		

2014 Dollars in Thousands

No	Item Description	Labor	Equipment	Disposal	Other	Contingency	Total
SR Pd 6	**Final Site Restoration and Lease Termination**						
Distributed							
19.01	Obtain Required Permits and Approvals	$404	$20	$0	$131	$139	$693
19.02	Install Temporary Structures	$6	$35	$0	$0	$6	$48
19.03	Procure Site Restoration Equipment	$0	$404	$0	$0	$101	$505
19.04	Install Temporary Seawall or Coffer Dam	$8,551	$17,624	$0	$0	$6,544	$32,718
19.05	Install Dewatering System and Effluent Treatment and Discharge Controls	$0	$0	$0	$1,427	$357	$1,784
19.06	Remove and Stockpile Existing Seawall Erosion Protection	$6	$11	$0	$0	$4	$21
19.07	Remove Unit 2 and 3 Seawall and Pedestrian Walkway	$3,206	$3,060	$4,558	$0	$2,706	$13,530
19.08	Remove Remaining Intake and Outfall Box Culvert	$336	$468	$2,188	$0	$748	$3,739
19.09	Remove Temporary Seawall or Coffer Dam	$11,791	$143	$0	$0	$2,983	$14,917
19.10	Backfill and Compaction of Excavation	$1,471	$2,238	$0	$0	$556	$4,265
19.11	Remove Dewatering System and Effluent Treatment	$0	$0	$0	$592	$148	$740
19.12	Install Shoreline Erosion Control and Restoration Features	$10	$144	$0	$0	$38	$192
19.13	Remove Railroad Tracks, Rails and Ballast	$63	$35	$0	$0	$24	$122
19.14	Remove Gunite Slope Protection	$262	$366	$1,710	$0	$585	$2,923
19.15	Remove Access Roads and Parking Lots	$240	$181	$0	$0	$105	$527
19.16	Finish Grading and Re-Vegetate Site	$27	$28	$0	$0	$14	$68
19.17	Remove Temporary Structures	$8	$7	$0	$0	$2	$18
Distributed	**Subtotal**	**$26,380**	**$24,763**	**$8,456**	**$2,151**	**$15,061**	**$76,810**
Undistributed							
3.01	Utility Staff	$2,219	$0	$0	$0	$555	$2,773
3.04	Insurance	$0	$0	$0	$605	$151	$756
3.05	Site Lease and Easement Expenses	$0	$0	$0	$507	$76	$583
3.06	Materials and Services	$0	$142	$0	$0	$35	$177
3.07	Energy	$0	$0	$0	$418	$104	$522
3.08	Decommissioning General Contractor Staff	$8,062	$0	$0	$0	$2,016	$10,078
3.09	Craft Worker Training	$504	$0	$0	$0	$126	$630
3.10	Workers Compensation Insurance	$0	$0	$0	$305	$76	$381
3.11	Severance	$0	$0	$0	$6,077	$1,519	$7,596
3.12	Property Tax	$0	$0	$0	$2,536	$634	$3,169
3.13	Utilities (Water, gas, phone)	$0	$31	$0	$0	$8	$38

Table 1
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Spent Fuel Alternative	Dry	Repository Opening Date:	1/1/2024		

2014 Dollars in Thousands

No	Item Description	Labor	Equipment	Disposal	Other	Contingency	Total
3.14	Tools and Equipment	$0	$24	$0	$0	$6	$31
Undistributed	**Subtotal**	**$10,785**	**$197**	**$0**	**$10,446**	**$5,307**	**$26,735**
SR Pd 6	**Subtotal**	**$37,165**	**$24,960**	**$8,456**	**$12,597**	**$20,367**	**$103,545**
C. Site Restoration	**Subtotal**	**$243,198**	**$181,428**	**$186,230**	**$233,951**	**$177,997**	**$1,022,804**
	Total	**$1,763,742**	**$505,191**	**$757,925**	**$566,786**	**$817,601**	**$4,411,246**

Appendix E

Annual Cash Flow Table

SONGS Annual Cost By Account

Prompt DECON Base Case, 2024 DOE Acceptance, Dry Storage

Unit No: Unit 2

2014 Dollars in Thousands

Year	License Termination	Spent Fuel	Site Restoration	Total
2013	$25,749	$63,891	$49,067	$138,706
2014	$79,799	$35,719	$15,089	$130,607
2015	$69,196	$106,308	$7,439	$182,943
2016	$54,541	$59,308	$3,730	$117,579
2017	$111,903	$59,308	$1,957	$173,168
2018	$47,520	$59,308	$0	$106,828
2019	$108,328	$27,554	$13,539	$149,420
2020	$185,482	$4,908	$36	$190,426
2021	$79,081	$4,908	$36	$84,026
2022	$54,785	$4,908	$1,927	$61,621
2023	$158,207	$4,908	$36	$163,151
2024	$37,930	$4,908	$16,848	$59,687
2025	$2,922	$4,908	$44,621	$52,451
2026	$2,922	$4,908	$19,412	$27,243
2027	$2,922	$4,908	$22,469	$30,299
2028	$2,922	$4,908	$31,688	$39,518
2029	$2,922	$4,908	$66,873	$74,704
2030	$2,922	$4,908	$71,867	$79,697
2031	$2,055	$5,089	$23,181	$30,325
2032	$2,122	$7,214	$0	$9,336
2033	$0	$7,214	$0	$7,214
2034	$0	$7,214	$0	$7,214
2035	$0	$7,228	$0	$7,228
2036	$0	$7,665	$0	$7,665
2037	$0	$7,665	$0	$7,665
2038	$0	$7,665	$0	$7,665
2039	$0	$7,665	$0	$7,665
2040	$0	$7,665	$0	$7,665
2041	$0	$7,665	$0	$7,665
2042	$0	$7,665	$0	$7,665
2043	$0	$7,665	$0	$7,665
2044	$0	$7,665	$0	$7,665
2045	$0	$7,665	$0	$7,665
2046	$0	$7,665	$0	$7,665
2047	$0	$7,665	$0	$7,665
2048	$0	$7,665	$0	$7,665

SONGS Annual Cost By Account

Prompt DECON Base Case, 2024 DOE Acceptance, Dry Storage

Unit No: Unit 2

2014 Dollars in Thousands

Year	License Termination	Spent Fuel	Site Restoration	Total
2049	$0	$7,667	$0	$7,667
2050	$0	$9,974	$20,177	$30,151
2051	$0	$6,573	$11,928	$18,500
2052	$0	$0	$1,377	$1,377
Total	$1,034,230	$623,209	$423,297	$2,080,735

SONGS Annual Cost By Account

Prompt DECON Base Case, 2024 DOE Acceptance, Dry Storage

Unit No: Unit 3

2014 Dollars in Thousands

Year	License Termination	Spent Fuel	Site Restoration	Total
2013	$26,566	$66,105	$49,067	$141,739
2014	$78,964	$40,156	$15,969	$135,089
2015	$74,096	$112,024	$9,390	$195,509
2016	$61,451	$64,405	$25,227	$151,083
2017	$40,631	$64,405	$3,799	$108,835
2018	$86,348	$64,405	$0	$150,753
2019	$96,521	$29,675	$13,908	$140,104
2020	$120,873	$4,908	$2,135	$127,916
2021	$194,090	$4,908	$575	$199,574
2022	$135,313	$4,908	$2,467	$142,688
2023	$114,581	$4,908	$1,511	$121,000
2024	$26,874	$4,908	$36,778	$68,560
2025	$2,922	$4,908	$40,655	$48,485
2026	$2,922	$4,908	$21,676	$29,507
2027	$2,922	$4,908	$25,848	$33,678
2028	$2,922	$4,908	$20,945	$28,776
2029	$2,922	$4,908	$117,321	$125,151
2030	$2,922	$4,908	$116,672	$124,503
2031	$2,055	$5,089	$25,501	$32,645
2032	$2,122	$7,214	$0	$9,336
2033	$0	$7,214	$0	$7,214
2034	$0	$7,214	$0	$7,214
2035	$0	$7,228	$0	$7,228
2036	$0	$7,665	$0	$7,665
2037	$0	$7,665	$0	$7,665
2038	$0	$7,665	$0	$7,665
2039	$0	$7,665	$0	$7,665
2040	$0	$7,665	$0	$7,665
2041	$0	$7,665	$0	$7,665
2042	$0	$7,665	$0	$7,665
2043	$0	$7,665	$0	$7,665
2044	$0	$7,665	$0	$7,665
2045	$0	$7,665	$0	$7,665
2046	$0	$7,665	$0	$7,665
2047	$0	$7,665	$0	$7,665
2048	$0	$7,665	$0	$7,665

SONGS Annual Cost By Account

Prompt DECON Base Case, 2024 DOE Acceptance, Dry Storage

Unit No: Unit 3

2014 Dollars in Thousands

Year	License Termination	Spent Fuel	Site Restoration	Total
2049	$0	$7,667	$0	$7,667
2050	$0	$9,974	$23,120	$33,094
2051	$0	$6,573	$45,566	$52,139
2052	$0	$0	$1,377	$1,377
Total	$1,078,016	$652,987	$599,507	$2,330,511

Appendix F

SDG&E SONGS Decommissioning Costs (100%)

Appendix F SDG&E SONGS Decommissioning Costs (100%)

San Diego Gas & Electric Company (SDG&E) provides the following information regarding its internal decommissioning costs, which it expects to incur and to fund on its own behalf (100%) in addition to its 20% share of the Decommissioning Cost Estimate.

I. BACKGROUND

As the 20% minority owner, SDG&E is contractually obligated to pay its 20% ownership share of decommissioning expenses for SONGS. These costs, outlined in the DCE, will be incurred by the decommissioning agent and SDG&E will receive invoicing for its proportional share.

II. SDG&E COSTS

Table F-1			
SDG&E SONGS DECOMMISSIONING COSTS (1,000's, $2014)			
Total Units 2 & 3	**SDG&E Labor**	**Other/ Non-Labor**	**Total Costs**
License Termination	$3,832	$1,047	$4,879
Spent Fuel Management	$2,729	$417	$3,147
Site Restoration	$1,904	$401	$2,305
Total	$8,465	$1,865	$10,330

In addition to SDG&E's 20% share of the costs outlined in the DCE, SDG&E also incurs internal costs related to its SONGS ownership. SDG&E incurs 100% of these Labor and Non-Labor costs related to SDG&E's oversight activities. These costs are apportioned into SCE's DCE categories of License Termination, Spent Fuel Management, and Site Restoration by determining the percentage of costs SCE allocated to each category and multiplying SDG&E's

costs by that same percentage for each category. SDG&E estimates that its total internal costs over the decommissioning period to be $10.33 million (2014$).

a. SDG&E LABOR

The first category, "SDG&E Labor" includes SDG&E staff who provide oversight of SONGS costs and activities. SDG&E's internal staffing efforts are expected to mirror site staffing where the three (3) full-time equivalents ("FTEs") are reduced after 2016 to two (2) FTEs, then to one (1) FTE after 2025, and eventually to zero (0) FTEs after 2032. After 2032, invoicing and oversight activities are anticipated to be minor during this period. Once ISFSI decommissioning is initiated on or around 2049, SDG&E plans to identify one (1) full-time equivalent through 2052.

These costs are shown in Table F-1 under the column heading of "SDG&E Labor" and are apportioned into SCE's categories of License Termination, Spent Fuel Management, and Site Restoration.

b. OTHER/NON-LABOR

The second type of SDG&E-specific costs are "Other/Non-Labor", which consist of outside decommissioning consultants and direct costs related to oversight activities.

To provide oversight of decommissioning activities, SDG&E has retained an external decommissioning consultant who has the expertise SDG&E requires. The external consultant is utilized to a greater extent through 2016 and then the consultant services are tapered off annually through 2025.

SDG&E also incurs direct costs related specifically to SDG&E's oversight activities at SONGS. These costs, which include travel reimbursement, phone services, training, and wireless

communication from SONGS, will coincide with the number of SDG&E SONGS oversight personnel FTEs.

These costs are shown in Table F-1 under the column heading of Other/Non-labor and are apportioned into SCE's categories of License Termination, Spent Fuel Management, and Site Restoration.

III. CONCLUSION

All of SDG&E's internal decommissioning costs presented in Table F-1 are separate and distinct from the costs incurred by the decommissioning agent and invoiced to SDG&E.

SDG&E will seek authority to access its nuclear decommissioning trust funds to pay for its proportional share of SONGS related decommissioning expenses and for its internal decommissioning costs incurred through a Commission-approved advice letter process consistent with the terms of the SDG&E Master Trust Agreement, and relevant rules and regulations of the Internal Revenue Service and the Nuclear Regulatory Commission.

SDG&E SONGS Detailed Annual Expenditures
Base Case:Prompt DECON, Time Reasonable Schedule, DOE Repository Opening 2024, Utility and DGC, Dry Storage
(2014 Dollars in Thousands)

Account Totals	Unit 2			Unit 3			Total			
	Labor	LLRW Burial	Other	Labor	LLRW Burial	Other	Labor	LLRW Burial	Other	Total
License Termination	$1,905	$0	$487	$1,927	$0	$560	$3,832	$0	$1,047	$4,879
Spent Fuel Management	$1,349	$0	$184	$1,380	$0	$233	$2,729	$0	$417	$3,147
Site Restoration	$761	$0	$153	$1,143	$0	$248	$1,904	$0	$401	$2,305
	$4,016	$0	$823	$4,450	$0	$1,041	$8,465	$0	$1,865	$10,330

Unit 2

	License Termination			Spent Fuel Management			Site Restoration			ISFSI D&D		
Year	Labor	LLRW Burial	Other	Labor	LLRW Burial	Other	Labor	LLRW Burial	Other	Labor	LLRW Burial	Other
2013	$25	$0	$0	$91	$0	$0	$0	$0	$1	$0	$0	$0
2014	$240	$0	$41	$122	$0	$22	$0	$0	$28	$0	$0	$0
2015	$225	$0	$20	$123	$0	$66	$6	$0	$5	$0	$0	$0
2016	$133	$0	$41	$203	$0	$22	$10	$0	$3	$0	$0	$0
2017	$88	$0	$77	$138	$0	$23	$4	$0	$2	$0	$0	$0
2018	$74	$0	$34	$144	$0	$24	$0	$0	$0	$0	$0	$0
2019	$162	$0	$47	$78	$0	$7	$0	$0	$21	$0	$0	$0
2020	$244	$0	$89	$11	$0	$1	$0	$0	$0	$0	$0	$0
2021	$146	$0	$35	$13	$0	$1	$0	$0	$0	$0	$0	$0
2022	$184	$0	$25	$20	$0	$1	$0	$0	$5	$0	$0	$0
2023	$206	$0	$57	$21	$0	$1	$0	$0	$0	$0	$0	$0
2024	$139	$0	$14	$29	$0	$1	$45	$0	$21	$0	$0	$0
2025	$6	$0	$1	$23	$0	$0	$89	$0	$9	$0	$0	$0
2026	$7	$0	$1	$30	$0	$0	$74	$0	$7	$0	$0	$0
2027	$7	$0	$1	$28	$0	$0	$73	$0	$7	$0	$0	$0
2028	$7	$0	$1	$27	$0	$0	$88	$0	$10	$0	$0	$0
2029	$5	$0	$0	$20	$0	$0	$71	$0	$6	$0	$0	$0
2030	$5	$0	$0	$21	$0	$0	$71	$0	$7	$0	$0	$0
2031	$3	$0	$1	$25	$0	$1	$82	$0	$7	$0	$0	$0
2032	$0	$0	$0	$0	$0	$0	$0	$0	$0	$0	$0	$0
2033	$0	$0	$0	$0	$0	$0	$0	$0	$0	$0	$0	$0
2034	$0	$0	$0	$0	$0	$0	$0	$0	$0	$0	$0	$0
2035	$0	$0	$0	$0	$0	$0	$0	$0	$0	$0	$0	$0
2036	$0	$0	$0	$0	$0	$0	$0	$0	$0	$0	$0	$0
2037	$0	$0	$0	$0	$0	$0	$0	$0	$0	$0	$0	$0
2038	$0	$0	$0	$0	$0	$0	$0	$0	$0	$0	$0	$0
2039	$0	$0	$0	$0	$0	$0	$0	$0	$0	$0	$0	$0
2040	$0	$0	$0	$0	$0	$0	$0	$0	$0	$0	$0	$0
2041	$0	$0	$0	$0	$0	$0	$0	$0	$0	$0	$0	$0
2042	$0	$0	$0	$0	$0	$0	$0	$0	$0	$0	$0	$0
2043	$0	$0	$0	$0	$0	$0	$0	$0	$0	$0	$0	$0
2044	$0	$0	$0	$0	$0	$0	$0	$0	$0	$0	$0	$0
2045	$0	$0	$0	$0	$0	$0	$0	$0	$0	$0	$0	$0
2046	$0	$0	$0	$0	$0	$0	$0	$0	$0	$0	$0	$0
2047	$0	$0	$0	$0	$0	$0	$0	$0	$0	$0	$0	$0
2048	$0	$0	$0	$0	$0	$0	$0	$0	$0	$0	$0	$0
2049	$0	$0	$0	$114	$0	$9	$0	$0	$0	$0	$0	$0
2050	$0	$0	$0	$46	$0	$2	$111	$0	$6	$0	$0	$0
2051	$0	$0	$0	$24	$0	$1	$35	$0	$4	$0	$0	$0
2052	$0	$0	$0	$0	$0	$0	$0	$0	$4	$0	$0	$0
	$1,905	$0	$487	$1,349	$0	$184	$761	$0	$153	$0	$0	$0

Unit 3

	License Termination			Spent Fuel Management			Site Restoration			ISFSI D&D		
Year	Labor	LLRW Burial	Other	Labor	LLRW Burial	Other	Labor	LLRW Burial	Other	Labor	LLRW Burial	Other
2013	$25	$0	$0	$91	$0	$0	$0	$0	$1	$0	$0	$0
2014	$235	$0	$41	$122	$0	$30	$2	$0	$28	$0	$0	$0
2015	$228	$0	$23	$123	$0	$71	$15	$0	$5	$0	$0	$0
2016	$134	$0	$53	$206	$0	$32	$34	$0	$38	$0	$0	$0
2017	$71	$0	$33	$139	$0	$34	$6	$0	$4	$0	$0	$0
2018	$83	$0	$67	$145	$0	$36	$0	$0	$0	$0	$0	$0
2019	$127	$0	$46	$79	$0	$10	$1	$0	$21	$0	$0	$0
2020	$174	$0	$50	$11	$0	$1	$5	$0	$0	$0	$0	$0
2021	$272	$0	$94	$13	$0	$1	$1	$0	$0	$0	$0	$0
2022	$220	$0	$83	$20	$0	$1	$1	$0	$6	$0	$0	$0
2023	$192	$0	$51	$21	$0	$1	$6	$0	$1	$0	$0	$0
2024	$125	$0	$13	$29	$0	$1	$79	$0	$51	$0	$0	$0
2025	$6	$0	$1	$23	$0	$0	$83	$0	$8	$0	$0	$0
2026	$7	$0	$1	$30	$0	$0	$80	$0	$8	$0	$0	$0
2027	$7	$0	$1	$28	$0	$0	$83	$0	$8	$0	$0	$0
2028	$7	$0	$1	$27	$0	$0	$73	$0	$5	$0	$0	$0
2029	$5	$0	$0	$20	$0	$0	$108	$0	$12	$0	$0	$0
2030	$5	$0	$0	$21	$0	$0	$105	$0	$11	$0	$0	$0
2031	$3	$0	$1	$25	$0	$1	$89	$0	$8	$0	$0	$0
2032	$0	$0	$0	$0	$0	$0	$0	$0	$0	$0	$0	$0
2033	$0	$0	$0	$0	$0	$0	$0	$0	$0	$0	$0	$0
2034	$0	$0	$0	$0	$0	$0	$0	$0	$0	$0	$0	$0
2035	$0	$0	$0	$0	$0	$0	$0	$0	$0	$0	$0	$0
2036	$0	$0	$0	$0	$0	$0	$0	$0	$0	$0	$0	$0
2037	$0	$0	$0	$0	$0	$0	$0	$0	$0	$0	$0	$0
2038	$0	$0	$0	$0	$0	$0	$0	$0	$0	$0	$0	$0
2039	$0	$0	$0	$0	$0	$0	$0	$0	$0	$0	$0	$0
2040	$0	$0	$0	$0	$0	$0	$0	$0	$0	$0	$0	$0
2041	$0	$0	$0	$0	$0	$0	$0	$0	$0	$0	$0	$0
2042	$0	$0	$0	$0	$0	$0	$0	$0	$0	$0	$0	$0
2043	$0	$0	$0	$0	$0	$0	$0	$0	$0	$0	$0	$0
2044	$0	$0	$0	$0	$0	$0	$0	$0	$0	$0	$0	$0
2045	$0	$0	$0	$0	$0	$0	$0	$0	$0	$0	$0	$0
2046	$0	$0	$0	$0	$0	$0	$0	$0	$0	$0	$0	$0
2047	$0	$0	$0	$0	$0	$0	$0	$0	$0	$0	$0	$0
2048	$0	$0	$0	$0	$0	$0	$0	$0	$0	$0	$0	$0
2049	$0	$0	$0	$114	$0	$9	$0	$0	$0	$0	$0	$0
2050	$0	$0	$0	$46	$0	$2	$69	$0	$8	$0	$0	$0
2051	$0	$0	$0	$24	$0	$1	$152	$0	$11	$0	$0	$0
2052	$0	$0	$0	$24	$0	$1	$152	$0	$11	$0	$0	$0
	$1,927	$0	$560	$1,380	$0	$233	$1,143	$0	$248	$0	$0	$0

SDG&E SONGS Detailed Annual Expenditures

Base Cas Prompt DECON, Time Reasonable Schedule, DOE Repository Opening 2024, Utility and DGC, Dry Storage
(2014 Dollars in Thousands)

Account Totals

	Unit 2			Unit 3			Total			
	Labor	LLRW Burial	Other	Labor	LLRW Burial	Other	Labor	LLRW Burial	Other	Total
License Termination	$1,905	$0	$487	$1,927	$0	$560	$3,832	$0	$1,047	$4,879
Spent Fuel Management	$1,349	$0	$184	$1,380	$0	$233	$2,729	$0	$417	$3,147
Site Restoration	$761	$0	$153	$1,143	$0	$248	$1,904	$0	$401	$2,305
	$4,016	**$0**	**$823**	**$4,450**	**$0**	**$1,041**	**$8,465**	**$0**	**$1,865**	**$10,330**

Unit 2 and 3 Total

	License Termination			Spent Fuel Management			Site Restoration			ISFSI D&D		
Year	Labor	LLRW Burial	Other	Labor	LLRW Burial	Other	Labor	LLRW Burial	Other	Labor	LLRW Burial	Other
2013	$50	$0	$0	$182	$0	$1	$0	$0	$3	$0	$0	$0
2014	$474	$0	$82	$243	$0	$51	$2	$0	$56	$0	$0	$0
2015	$453	$0	$42	$245	$0	$137	$21	$0	$10	$0	$0	$0
2016	$267	$0	$95	$409	$0	$53	$44	$0	$41	$0	$0	$0
2017	$159	$0	$109	$277	$0	$57	$10	$0	$5	$0	$0	$0
2018	$157	$0	$101	$289	$0	$60	$0	$0	$0	$0	$0	$0
2019	$288	$0	$93	$157	$0	$16	$1	$0	$42	$0	$0	$0
2020	$418	$0	$139	$23	$0	$2	$5	$0	$0	$0	$0	$0
2021	$419	$0	$129	$26	$0	$2	$1	$0	$0	$0	$0	$0
2022	$404	$0	$107	$41	$0	$3	$2	$0	$11	$0	$0	$0
2023	$399	$0	$108	$42	$0	$3	$6	$0	$1	$0	$0	$0
2024	$264	$0	$27	$58	$0	$2	$124	$0	$72	$0	$0	$0
2025	$11	$0	$1	$45	$0	$0	$171	$0	$17	$0	$0	$0
2026	$15	$0	$3	$60	$0	$1	$153	$0	$15	$0	$0	$0
2027	$14	$0	$2	$57	$0	$1	$157	$0	$15	$0	$0	$0
2028	$13	$0	$2	$53	$0	$1	$161	$0	$16	$0	$0	$0
2029	$10	$0	$0	$39	$0	$0	$179	$0	$18	$0	$0	$0
2030	$10	$0	$0	$42	$0	$0	$176	$0	$18	$0	$0	$0
2031	$6	$0	$2	$50	$0	$1	$172	$0	$15	$0	$0	$0
2032	$0	$0	$0	$0	$0	$0	$0	$0	$0	$0	$0	$0
2033	$0	$0	$0	$0	$0	$0	$0	$0	$0	$0	$0	$0
2034	$0	$0	$0	$0	$0	$0	$0	$0	$0	$0	$0	$0
2035	$0	$0	$0	$0	$0	$0	$0	$0	$0	$0	$0	$0
2036	$0	$0	$0	$0	$0	$0	$0	$0	$0	$0	$0	$0
2037	$0	$0	$0	$0	$0	$0	$0	$0	$0	$0	$0	$0
2038	$0	$0	$0	$0	$0	$0	$0	$0	$0	$0	$0	$0
2039	$0	$0	$0	$0	$0	$0	$0	$0	$0	$0	$0	$0
2040	$0	$0	$0	$0	$0	$0	$0	$0	$0	$0	$0	$0
2041	$0	$0	$0	$0	$0	$0	$0	$0	$0	$0	$0	$0
2042	$0	$0	$0	$0	$0	$0	$0	$0	$0	$0	$0	$0
2043	$0	$0	$0	$0	$0	$0	$0	$0	$0	$0	$0	$0
2044	$0	$0	$0	$0	$0	$0	$0	$0	$0	$0	$0	$0
2045	$0	$0	$0	$0	$0	$0	$0	$0	$0	$0	$0	$0
2046	$0	$0	$0	$0	$0	$0	$0	$0	$0	$0	$0	$0
2047	$0	$0	$0	$0	$0	$0	$0	$0	$0	$0	$0	$0
2048	$0	$0	$0	$0	$0	$0	$0	$0	$0	$0	$0	$0
2049	$0	$0	$0	$228	$0	$18	$0	$0	$0	$0	$0	$0
2050	$0	$0	$0	$93	$0	$4	$180	$0	$14	$0	$0	$0
2051	$0	$0	$0	$48	$0	$3	$187	$0	$16	$0	$0	$0
2052	$0	$0	$0	$24	$0	$1	$152	$0	$16	$0	$0	$0
	$3,832	**$0**	**$1,047**	**$2,729**	**$0**	**$417**	**$1,904**	**$0**	**$401**	**$0**	**$0**	**$0**

Unit 2 and 3 Project Totals

Year	License Term	Spent Fuel	Site Restoration	Total
2013	$51	$182	$3	$236
2014	$557	$294	$58	$909
2015	$496	$382	$31	$909
2016	$362	$462	$85	$909
2017	$269	$334	$15	$618
2018	$258	$349	$0	$608
2019	$381	$173	$43	$598
2020	$557	$25	$5	$588
2021	$548	$28	$2	$578
2022	$511	$43	$13	$567
2023	$506	$44	$7	$557
2024	$291	$60	$196	$547
2025	$12	$45	$188	$246
2026	$18	$60	$168	$246
2027	$16	$57	$172	$246
2028	$15	$54	$177	$246
2029	$10	$39	$197	$246
2030	$11	$42	$194	$246
2031	$9	$51	$187	$246
2032	$0	$0	$0	$0
2033	$0	$0	$0	$0
2034	$0	$0	$0	$0
2035	$0	$0	$0	$0
2036	$0	$0	$0	$0
2037	$0	$0	$0	$0
2038	$0	$0	$0	$0
2039	$0	$0	$0	$0
2040	$0	$0	$0	$0
2041	$0	$0	$0	$0
2042	$0	$0	$0	$0
2043	$0	$0	$0	$0
2044	$0	$0	$0	$0
2045	$0	$0	$0	$0
2046	$0	$0	$0	$0
2047	$0	$0	$0	$0
2048	$0	$0	$0	$0
2049	$0	$246	$0	$246
2050	$0	$97	$195	$291
2051	$0	$51	$203	$253
2052	$0	$25	$168	$193
	$4,879	**$3,147**	**$2,305**	**$10,330**

UNITED STATES NUCLEAR REGULATORY COMMISSION

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

July 17, 2015

Mr. Thomas J. Palmisano
Vice President and Chief Nuclear Officer
Southern California Edison Company
San Onofre Nuclear Generating Station
P.O. Box 128
San Clemente, CA 92674-0128

SUBJECT: SAN ONOFRE NUCLEAR GENERATING STATION, UNITS 2 AND 3 - ISSUANCE OF AMENDMENT FOR PERMANENTLY SHUTDOWN AND DEFUELED OPERATING LICENSE AND TECHNICAL SPECIFICATIONS (TAC NOS. MF3774 AND MF3775)

Dear Mr. Palmisano:

The U.S. Nuclear Regulatory Commission has issued the enclosed Amendment No. 230 to Facility Operating License No. NPF-10, and Amendment No. 223 to Facility Operating License No. NPF-15, for the San Onofre Nuclear Generating Station (SONGS), Units 2 and 3, respectively. The amendments consist of changes to the SONGS facility operating licenses and the Technical Specifications (TSs) in response to your application dated March 21, 2014, as supplemented by letters dated October 1, 2014; and February 23, February 25, and March 18, 2015.

The proposed amendments revise the operating licenses and associated TSs to reflect the permanent cessation of reactor operations and the permanently defueled condition of the reactor vessels at SONGS Units 2 and 3. In general, the changes eliminate those TSs applicable in operating MODES; MODES where fuel is emplaced in the reactor vessel, and certain TSs required for movement of irradiated fuel assemblies. Changes were also made to the TS definitions, administrative controls, and related to programs and procedures. The proposed amendments also revise the facility operating licenses to clarify or remove certain conditions no longer relevant and add conditions consistent with other permanently shutdown and defueled reactors. Related Amendment Nos. 227 and 220 for SONGS Units 2 and 3, respectively, were issued on September 30, 2014, to revise and remove certain requirements from Section 5.0, "Administrative Controls," of the SONGS Units 2 and 3 TSs to reflect the permanently shutdown and defueled staffing and training requirements for SONGS Units 2 and 3 operations staff.

T. Palmisano

A copy of the Safety Evaluation is also enclosed. The Notice of Issuance will be included in the Commission's biweekly *Federal Register* notice.

Sincerely,

Thomas J. Wengert, Senior Project Manager
Plant Licensing IV-2 and Decommissioning
Transition Branch
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-361 and 50-362

Enclosures:
1. Amendment No. 230 to NPF-10
2. Amendment No. 223 to NPF-15
3. Safety Evaluation

cc w/enclosures: Distribution via Listserv

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SOUTHERN CALIFORNIA EDISON COMPANY

SAN DIEGO GAS AND ELECTRIC COMPANY

THE CITY OF RIVERSIDE, CALIFORNIA

DOCKET NO. 50-361

SAN ONOFRE NUCLEAR GENERATING STATION, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 230
License No. NPF-10

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Southern California Edison Company, et al. (SCE or the licensee), dated March 21, 2014, as supplemented by letters dated October 1, 2014; and February 23, February 25, and March 18, 2015, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

Enclosure 1

Case: 20-70899, 03/31/2020, ID: 11647799, DktEntry: 2-3, Page 291 of 301

2. Accordingly, Facility Operating License No. NPF-10 is hereby amended to read, as follows, as indicated in the attachment to this license amendment.

Paragraph 2.B.(2) of Facility Operating License No. NPF-10 is hereby amended to read, as follows:

(2) Southern California Edison Company (SCE), pursuant to Section 103 of the Act and 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities", to possess and use the facility at the designated location in San Diego County, California, in accordance with the procedures and limitations set forth in this license;

Paragraph 2.B.(3) of Facility Operating License No. NPF-10 is hereby amended to read, as follows:

(3) SCE, pursuant to the Act and 10 CFR Part 70, to possess at any time special nuclear material that was used as reactor fuel, in accordance with the limitations for storage, as described in the Final Safety Analysis Report, as supplemented and amended;

Paragraph 2.B.(4) of Facility Operating License No. NPF-10 is hereby amended to read, as follows:

(4) SCE, pursuant to the Act and 10 CFR Parts 30, 40, and 70, to receive, possess, and use at any time any byproduct, source and special nuclear material as sealed sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required; and possess any byproduct, source and special material as sealed neutron sources that was used for reactor startup;

Paragraph 2.C.(1) of Facility Operating License No. NPF-10 is hereby amended to read, as follows:

(1) Deleted

Paragraph 2.C.(2) of Facility Operating License No. NPF-10 is hereby amended to read, as follows:

(2) <u>Technical Specifications</u>

The Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, as revised through Amendment No. 230, are hereby incorporated in the license. Southern California Edison Company shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

Paragraph 2.C.(14) of Facility Operating License No. NPF-10 is hereby amended to read, as follows:

(14) Deleted

Paragraph 2.C.(27) of Facility Operating License No. NPF-10 is hereby amended to read, as follows:

(27) Deleted

New License Condition 2.C.(28) of Facility Operating License No. NPF-10 is hereby added to read, as follows:

(28) Prior to February 16, 2021, if all spent fuel has not been removed from the Unit 2 spent fuel pool, an aging-management program shall be submitted for NRC approval. The scope of the program shall include those long-lived, passive structures and components that are needed to provide reasonable assurance of the safe condition of the spent fuel in the spent fuel pool. Once approved, the program shall be described in the Updated Final Safety Analysis Report and shall remain in effect for Unit 2 until such time that all spent fuel has been removed from the Unit 2 spent fuel pool.

Paragraph 2.J of Facility Operating License No. NPF-10 is hereby amended to read, as follows:

J. Deleted

New License Condition 3 of Facility Operating License No. NPF-10 is hereby added to read, as follows:

3. On June 12, 2013, Southern California Edison (SCE) certified that operations at San Onofre Nuclear Generating Station Unit 2 would permanently cease in accordance with 10 CFR 50.82(a)(1)(i). On July 22, 2013, SCE certified that the fuel had been permanently removed from the reactor vessel in accordance with 10 CFR 50.82(a)(1)(ii). As a result, the 10 CFR 50 license no longer authorizes operation of the reactor, or the emplacement or retention of fuel in the reactor vessel.

This license is effective as of the date of issuance and authorizes ownership and possession of San Onofre Nuclear Generating Station Unit 2 until the Commission notifies the licensee in writing that the license is terminated. The licensee shall:

A. Take actions necessary to decommission the plant and continue to maintain the facility, including, where applicable, the storage, control and maintenance of the spent fuel, in a safe condition; and

B. Conduct activities in accordance with all other restrictions applicable to the facility in accordance with the NRC regulations and the applicable provisions of the 10 CFR 50 facility license as defined in Section 2 of this license.

3. This license amendment is effective as of the date of its issuance and shall be implemented within 60 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Meena K. Khanna, Chief
Plant Licensing IV-2 and Decommissioning
Transition Branch
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Facility
Operating License No. NPF-10
and Technical Specifications

Date of Issuance: July 17, 2015

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SOUTHERN CALIFORNIA EDISON COMPANY

SAN DIEGO GAS AND ELECTRIC COMPANY

THE CITY OF RIVERSIDE, CALIFORNIA

DOCKET NO. 50-362

SAN ONOFRE NUCLEAR GENERATING STATION, UNIT 3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 223
License No. NPF-15

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Southern California Edison Company, et al. (SCE or the licensee), dated March 21, 2014, as supplemented by letters dated October 1, 2014; and February 23, February 25, and March 18, 2015, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

Enclosure 2

2. Accordingly, Facility Operating License No. NPF-15 is hereby amended to read, as follows, as indicated in the attachment to this license amendment.

Paragraph 2.B.(2) of Facility Operating License No. NPF-15 is hereby amended to read, as follows:

(2) Southern California Edison Company (SCE), pursuant to Section 103 of the Act and 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities", to possess and use the facility at the designated location in San Diego County, California, in accordance with the procedures and limitations set forth in this license;

Paragraph 2.B.(3) of Facility Operating License No. NPF-15 is hereby amended to read, as follows:

(3) SCE, pursuant to the Act and 10 CFR Part 70, to possess at any time special nuclear material that was used as reactor fuel, in accordance with the limitations for storage, as described in the Final Safety Analysis Report, as supplemented and amended;

Paragraph 2.B.(4) of Facility Operating License No. NPF-15 is hereby amended to read, as follows:

(4) SCE, pursuant to the Act and 10 CFR Parts 30, 40, and 70, to receive, possess, and use at any time any byproduct, source and special nuclear material as sealed sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required; and possess any byproduct, source and special material as sealed neutron sources that was used for reactor startup;

Paragraph 2.C.(1) of Facility Operating License No. NPF-15 is hereby amended to read, as follows:

(1) Deleted

Paragraph 2.C.(2) of Facility Operating License No. NPF-15 is hereby amended to read, as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, as revised through Amendment No. 223, are hereby incorporated in the license. Southern California Edison Company shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

Paragraph 2.C.(12) of Facility Operating License No. NPF-15 is hereby amended to read, as follows:

(12) Deleted

Paragraph 2.C.(28) of Facility Operating License No. NPF-15 is hereby amended to read, as follows:

(28) Deleted

New License Condition 2.C.(29) of Facility Operating License No. NPF-15 is hereby added to read, as follows:

(29) Prior to February 16, 2021, if all spent fuel has not been removed from the Unit 3 spent fuel pool, an aging-management program shall be submitted for NRC approval. The scope of the program shall include those long-lived, passive structures and components that are needed to provide reasonable assurance of the safe condition of the spent fuel in the spent fuel pool. Once approved, the program shall be described in the Updated Final Safety Analysis Report and shall remain in effect for Unit 3 until such time that all spent fuel has been removed from the Unit 3 spent fuel pool.

Paragraph 2.J of Facility Operating License No. NPF-15 is hereby amended to read, as follows:

J. Deleted

New License Condition 3 of Facility Operating License No. NPF-15 is hereby added to read, as follows:

3. On June 12, 2013, Southern California Edison (SCE) certified that operations at San Onofre Nuclear Generating Station Unit 3 would permanently cease in accordance with 10 CFR 50.82(a)(1)(i). On June 28, 2013, SCE certified that the fuel had been permanently removed from the reactor vessel in accordance with 10 CFR 50.82(a)(1)(ii). As a result, the 10 CFR 50 license no longer authorizes operation of the reactor, or the emplacement or retention of fuel in the reactor vessel.

This license is effective as of the date of issuance and authorizes ownership and possession of San Onofre Nuclear Generating Station Unit 3 until the Commission notifies the licensee in writing that the license is terminated. The licensee shall:

A. Take actions necessary to decommission the plant and continue to maintain the facility, including, where applicable, the storage, control and maintenance of the spent fuel, in a safe condition; and

B. Conduct activities in accordance with all other restrictions applicable to the facility in accordance with the NRC regulations and the applicable provisions of the 10 CFR 50 facility license as defined in Section 2 of this license.

3. This license amendment is effective as of the date of its issuance and shall be implemented within 60 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Meena K. Khanna, Chief
Plant Licensing IV-2 and Decommissioning
Transition Branch
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Facility
Operating License No. NPF-15
and Technical Specifications

Date of Issuance: July 17, 2015

Case: 20-70899, 03/31/2020, ID: 11647799, DktEntry: 2-3, Page 298 of 301

ATTACHMENT TO LICENSE AMENDMENT NO. 230

TO FACILITY OPERATING LICENSE NO. NPF-10

AND LICENSE AMENDMENT NO. 223

TO FACILITY OPERATING LICENSE NO. NPF-15

DOCKET NOS. 50-361 AND 50-362

Replace the following pages of the Facility Operating License Nos. NPF-10 and NPF-15, and Appendix A Technical Specifications, with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

Facility Operating License No. NPF-10

Remove	Insert
-2-	-2-
-3-	-3-
-5-	-5-
-8-	-8-
-9-	-9-
---	-10-

Facility Operating License No. NPF-15

Remove	Insert
-2-	-2-
-3-	-3-
-4-	-4-
-5-	-5-
-7-	-7-
-9-	-9-

Technical Specifications

Remove	Insert
All pages	All pages

Facility Operating License No. NPF-10

Revised License Pages

Case: 20-70899, 03/31/2020, ID: 11647799, DktEntry: 2-3, Page 300 of 301

G. The licensees have satisfied the applicable provisions of 10 CFR Part 140, "Financial Protection Requirements and Indemnity Agreements," of the Commission's regulations;

H. The issuance of this license will not be inimical to the common defense and security or to the health and safety of the public;

I. After weighing the environmental, economic, technical, and other benefits of the facility against environmental and other costs and considering available alternatives, the issuance of Facility Operating License No. NPF-10, subject to the condition for protection of the environment set forth herein, is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied; and

J. The receipt, possession, and use of source, byproduct, and special nuclear material as authorized by this license will be in accordance with the Commission's regulations in 10 CFR Parts 30, 40 and 70.

2. Based on the foregoing findings and the Partial Initial Decision issued by the Atomic Safety and Licensing Board on January 11, 1982, regarding this facility, Facility Operating License No. NPF-10 is hereby issued to the Southern California Edison Company, the San Diego Gas and Electric Company, the City of Riverside, California, and the City of Anaheim, California[1] to read as follows:

A. This license applies to the San Onofre Nuclear Generating Station, Unit 2, a pressurized water nuclear reactor and associated equipment (the facility), owned by the licensees. The facility is located in San Diego County, California, and is described in the Final Safety Analysis Report as supplemented and amended, and the Environmental Report, as supplemented and amended.

B. Subject to the conditions and requirements incorporated herein, the Commission hereby licenses:

(1) Southern California Edison Company, San Diego Gas and Electric Company, the City of Riverside, California, and the City of Anaheim, California[1] to possess the facility at the designated location in San Diego County, California, in accordance with the procedures and limitations set forth in this license;

(2) Southern California Edison Company (SCE), pursuant to Section 103 of the Act and 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," to possess and use the facility at the designated location in San Diego County, California in accordance with the procedures and limitations set forth in this license;

---

[1]The City of Anaheim has transferred its ownership interests in the facility, and entitlement to facility output, to Southern California Edison Company, except that it retains its ownership interests in its spent nuclear fuel and the facility's independent spent fuel storage installation located on the facility's site. In addition, the City of Anaheim retains financial responsibility for its spent fuel and for a portion of the facility's decommissioning costs. The City of Anaheim remains a licensee for purposes of its retained interests and liabilities.

(3) SCE, pursuant to the Act and 10 CFR Part 70, to possess at any time special nuclear material that was used as reactor fuel, in accordance with the limitations for storage, as described in the Final Safety Analysis Report, as supplemented and amended;

(4) SCE, pursuant to the Act and 10 CFR Parts 30, 40, and 70, to receive, possess, and use at any time any byproduct, source and special nuclear material as sealed sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required; and possess any byproduct, source and special material as sealed neutron sources that was used for reactor startup;

(5) SCE, pursuant to the Act and 10 CFR Parts 30, 40, and 70 to receive, possess, and use in amounts as required any byproduct, source or special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactive apparatus or components; and

(6) SCE, pursuant to the Act and 10 CFR Parts 30, 40, and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of San Onofre Nuclear Generating Station, Units 1 and 2 and by the decommissioning of San Onofre Nuclear Generating Station Unit 1.

C. This license shall be deemed to contain and is subject to the conditions specified in the Commission's regulations set forth in 10 CFR Chapter I and is subject to all applicable provisions of the Act and to the rules, regulations and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) Deleted

(2) <u>Technical Specifications</u>

The Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, as revised through Amendment No. 230, are hereby incorporated in the license. Southern California Edison Company shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

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(14) Deleted

(15) Turbine Disc Inspection (Section 10.2.2, SER)

Deleted by Amendment No. 185

(16) Radioactive Waste System (Section 11.1, SER, SSER #5)

Deleted by Amendment No. 185

(17) Purge System Monitors (Section 11.3, SER, SSER #5)

Deleted by Amendment No. 185

(18) Initial Test Program (Section 14, SER)

Deleted by Amendment No. 185

(19) NUREG-0737 Conditions (Section 22)

a. Shift Technical Advisor (I.A.1.1, SSER #1)
Deleted by Amendment No. 185

b. Shift Manning (I.A.1.3. SSER #1, SSER #5)
Deleted by Amendment No. 147

c. Independent Safety Engineering Group (1.B.1.2. SSER #1)

Deleted by Amendment No. 185

d. Procedures for Transients and Accidents (I.C.1, SSER #1, SSER #2, SSER #5)

Deleted by Amendment No. 185

6. Training on integrated fire response strategy
7. Spent fuel pool mitigation measures

(c) Actions to minimize release to include consideration of:
1. Water spray scrubbing
2. Dose to onsite responders

(27) Deleted

(28) Prior to February 16, 2021, if all spent fuel has not been removed from the Unit 2 spent fuel pool, an aging-management program shall be submitted for NRC approval. The scope of the program shall include those long-lived, passive structures and components that are needed to provide reasonable assurance of the safe condition of the spent fuel in the spent fuel pool. Once approved, the program shall be described in the Updated Final Safety Analysis Report and shall remain in effect for Unit 2 until such time that all spent fuel has been removed from the Unit 2 spent fuel pool.

D. Exemptions to certain requirements of Appendices G, H and J to 10 CFR Part 50 are described in the Office of Nuclear Reactor Regulation's Safety Evaluation Report. These exemptions are authorized by law and will not endanger life or property or the common defense and security and are otherwise in the public interest. Therefore, these exemptions are hereby granted. The facility will operate, to the extent authorized herein, in conformity with the application, as amended, the provisions of the Act, and the regulations of the Commission.

E. SCE shall fully implement and maintain in effect all provisions of the Commission- approved physical security, training and qualification, and safeguards contingency plans including amendments made pursuant to provisions of the Miscellaneous Amendments and Search Requirements revisions to 10 CFR 73.55 (51 FR 27817 and 27822) and to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The combined set of plans, which contain Safeguards Information protected under 10 CFR 73.21 is entitled: "San Onofre Nuclear Generating Station Security, Training and Qualification, and Safeguards Contingency Plan, Revision 2" submitted by letter dated May 15, 2006. SCE shall fully implement and maintain in effect all provisions of the Commission-approved cyber security plan (CSP), including changes made pursuant to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The SONGS CSP was approved by License Amendment No. 225.

F. This license is subject to the following additional condition for the protection of the environment:

Before engaging in activities that may result in a significant adverse environmental impact that was not evaluated or that is significantly greater than that evaluated in the Final Environmental Statement, SCE shall provide a written notification of such activities to the NRC Office of Nuclear Reactor Regulation and receive written approval from that office before proceeding with such activities.

G. DELETED

H. SCE shall notify the Commission, as soon as possible but not later than one hour, of any accident at this facility which could result in an unplanned release of quantities of fission products in excess of allowable limits for normal operation established by the Commission.

I. SCE shall have and maintain financial protection of such type and in such amounts as the Commission shall require in accordance with Section 170 of the Atomic Energy Act of 1954, as amended, to cover public liability claims.

J. Deleted

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*On September 29, 1983, the Safeguards Contingency Plan was made a separate, companion document to the Physical Security Plan pursuant to the authority of 10 CFR 50.54.

3. On June 12, 2013, Southern California Edison (SCE) certified that operations at San Onofre Nuclear Generating Station Unit 2 would permanently cease in accordance with 10 CFR 50.82(a)(1)(i). On July 22, 2013, SCE certified that the fuel had been permanently removed from the reactor vessel in accordance with 10 CFR 50.82(a)(1)(ii). As a result, the 10 CFR 50 license no longer authorizes operation of the reactor, or the emplacement or retention of fuel in the reactor vessel.

This license is effective as of the date of issuance and authorizes ownership and possession of San Onofre Nuclear Generating Station Unit 2 until the Commission notifies the licensee in writing that the license is terminated. The licensee shall:

A. Take actions necessary to decommission the plant and continue to maintain the facility, including, where applicable, the storage, control and maintenance of the spent fuel, in a safe condition; and

B. Conduct activities in accordance with all other restrictions applicable to the facility in accordance with the NRC regulations and the applicable provisions of the 10 CFR 50 facility license as defined in Section 2 of this license.

FOR THE NUCLEAR REGULATORY COMMISSION

Original Signed by
Harold R. Denton

Harold R. Denton, Director
Office of Nuclear Reactor Regulation

Enclosures:
1. Appendix A (Technical Specifications)
2. Appendix B (Environmental Protection Plan)
3. Appendix C (Antitrust Conditions)

Date of Issuance: FEB 16 1982

Facility Operating License No. NPF-15

Revised License Pages

F. The licensees have satisfied the applicable provisions of 10 CFR Part 140, "Financial Protection Requirements and Indemnity Agreements," of the Commission's regulations;

G. The issuance of this license will not be inimical to the common defense and security or to the health and safety of the public;

H. After weighing the environmental, economic, technical, and other benefits of the facility against environmental and other costs and considering available alternatives, the issuance of Facility Operating License No. NPF-15, subject to the condition for protection of the environment set forth herein, is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied; and

I. The receipt, possession, and use of source, byproduct, and special nuclear material as authorized by this license will be in accordance with the Commission's regulations in 10 CFR Parts 30, 40 and 70.

2. Based on the foregoing findings and the Partial Initial Decision issued by the Atomic Safety and Licensing Board on January 11, 1982, and the Initial Decision issued by the Atomic Safety and Licensing Board on May 14, 1982 regarding this facility, Facility Operating License No. NPF-15 is hereby issued to the Southern California Edison Company, the San Diego Gas and Electric Company, the City of Riverside, California, and the City of Anaheim, California[1] to read as follows:

A. This license applies to the San Onofre Nuclear Generating Station, Unit 3, a pressurized water nuclear reactor and associated equipment (the facility), owned by the licensees. The facility is located in San Diego County, California, and is described in the Final Safety Analysis Report, as amended, through Amendment 30, and the Environmental Report, as amended, through Amendment 6.

B. Subject to the conditions and requirements incorporated herein, the Commission hereby licenses:

(1) Southern California Edison Company, San Diego Gas and Electric Company, the City of Riverside, California, and the City of Anaheim, California[1] to possess the facility at the designated location in San Diego County, California, in accordance with the procedures and limitations set forth in this license;

(2) Southern California Edison Company (SCE), pursuant to Section 103 of the Act and 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," to possess and use the facility at the designated location in San Diego County, California in accordance with the procedures and limitations set forth in this license;

[1]The City of Anaheim has transferred its ownership interests in the facility, and entitlement to facility output, to Southern California Edison Company, except that it retains its ownership interests in its spent nuclear fuel and the facility's independent spent fuel storage installation located on the facility's site. In addition, the City of Anaheim retains financial responsibility for its spent fuel and for a portion of the facility's decommissioning costs. The City of Anaheim remains a licensee for purposes of its retained interests and liabilities.

(3) SCE, pursuant to the Act and 10 CFR Part 70, to possess at any time special nuclear material that was used as reactor fuel, in accordance with the limitations for storage, as described in the Final Safety Analysis Report, as supplemented and amended;

(4) SCE, pursuant to the Act and 10 CFR Parts 30, 40, and 70, to receive, possess, and use at any time any byproduct, source and special nuclear material as sealed sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required; and possess any byproduct, source and special material as sealed neutron sources that was used for reactor startup;

(5) SCE, pursuant to the Act and 10 CFR Parts 30, 40, and 70 to receive, possess, and use in amounts as required any byproduct, source or special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactive apparatus or components; and

(6) SCE, pursuant to the Act and 10 CFR Parts 30, 40, and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of San Onofre Nuclear Generating Station, Units 1 and 3 and by the decommissioning of San Onofre Nuclear Generating Station Unit 1.

C. This license shall be deemed to contain and is subject to the conditions specified in the Commission's regulations set forth in 10 CFR Chapter I and is subject to all applicable provisions of the Act and to the rules, regulations and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) Deleted

(2) <u>Technical Specifications</u>

The Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, as revised through Amendment No. 223, are hereby incorporated in the license. Southern California Edison Company shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

(3) Antitrust Conditions

SCE shall comply with the antitrust conditions delineated in Appendix C to this license.

(4) Containment Tendon Surveillance (Section *3.8.1, SER, SSER #5)

Deleted by Amendment No. 26

(5) Environmental Qualification (Section 3.11, SER, SSER #3, SSER #4)

Deleted by Amendment No. 49

(6) High Burnup Fission Gas Release (Section 4.2.2.2, SER)

Deleted by Amendment No. 176

(7) Low Temperature Overpressurization Protection (Section 5.2.2.2, SER)

Deleted by Amendment No. 176

(8) Volume Control Tank Control Logic (Section 7.3.5, SSER #4)

Deleted by Amendment No. 176

(9) Compliance with Regulatory Guide 1.97 (Section 7.5.1, SER, SSER #5)

Deleted by Amendment No. 176

(10) Control System Failures (Section 7.7, SER, SSER #4)

Deleted by Amendment No. 176

(11) Diesel Generator Modifications (Section 8.3.1, SER)

Deleted by Amendment No. 176

(12) Deleted

*The parenthetical notation following the title of many license conditions denotes the section of the Safety Evaluation Report and/or its supplements wherein the license condition is discussed.

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(13) Turbine Disc Inspection (Section 10.2.2, SER)

Deleted by Amendment No. 176

(14) Radioactive Waste System (Section 11.1, SER, SSER #5)

Deleted by Amendment No. 176

(15) Purge System Monitors (Section 11.3, SER, SSER #5)

Deleted by Amendment No. 176

(16) Initial Test Program (Section 14, SER)

Deleted by Amendment No. 176

(17) NUREG-0737 Conditions (Section 22)

Deleted by Amendment No. 176

a. Procedures for Transients and Accidents (I.C.1, SSER #1, SSER #2, SSER #5)

Deleted by Amendment No. 176

b. Procedures for Verifying Correct Performance of Operating Activities (I.C.6, SSER #1)

Deleted by Amendment No. 176

c. Control Room Design Review (I.D.1, SSER #1)

Deleted by Amendment No. 176

d. Post Accident Sampling System (NUREG-0737 Item II.B.3)

Deleted by Amendment No. 169

e. Direct Indication of Safety Valve Position (II.D.3, SSER #1)

Deleted by Amendment No. 176

f. AFW Pump 48-hour Endurance Test (II.E.1.1, SSER #11)

Deleted by Amendment No. 176

g. Emergency Power Supply for Pressurizer Heaters (II.E.3.1, SSER #1, SSER #5)

Deleted by Amendment No. 176

h. ICC Instrumentation (II.F.2, SSER #1, SSER #2, SSER #4)

Deleted by Amendment No. 176

(27) Mitigation Strategy License Condition

Develop and maintain strategies for addressing large fires and explosions and that include the following key areas:

(a) Fire fighting response strategy with the following elements:
1. Pre-defined coordinated fire response strategy and guidance
2. Assessment of mutual aid fire fighting assets
3. Designated staging areas for equipment and materials
4. Command and control
5. Training of response personnel

(b) Operations to mitigate fuel damage considering the following:
1. Protection and use of personnel assets
2. Communications
3. Minimizing fire spread
4. Procedures for implementing integrated fire response strategy
5. Identification of readily-available pre-staged equipment
6. Training on integrated fire response strategy
7. Spent fuel pool mitigation measures

(c) Actions to minimize release to include consideration of:
1. Water spray scrubbing
2. Dose to onsite responders

(28) Deleted

(29) Prior to February 16, 2021, if all spent fuel has not been removed from the Unit 3 spent fuel pool, an aging-management program shall be submitted for NRC approval. The scope of the program shall include those long-lived, passive structures and components that are needed to provide reasonable assurance of the safe condition of the spent fuel in the spent fuel pool. Once approved, the program shall be described in the Updated Final Safety Analysis Report and shall remain in effect for Unit 3 until such time that all spent fuel has been removed from the Unit 3 spent fuel pool.

D. Exemptions to certain requirements of Appendices G, H and J to 10 CFR Part 50 are described in the Office of Nuclear Reactor Regulation's Safety Evaluation

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J. Deleted

K. Deleted by Amendment No. 176

3. On June 12, 2013, Southern California Edison (SCE) certified that operations at San Onofre Nuclear Generating Station Unit 3 would permanently cease in accordance with 10 CFR 50.82(a)(1)(i). On June 28, 2013, SCE certified that the fuel had been permanently removed from the reactor vessel in accordance with 10 CFR 50.82(a)(1)(ii). As a result, the 10 CFR 50 license no longer authorizes operation of the reactor, or the emplacement or retention of fuel in the reactor vessel.

This license is effective as of the date of issuance and authorizes ownership and possession of San Onofre Nuclear Generating Station Unit 3 until the Commission notifies the licensee in writing that the license is terminated. The licensee shall:

A. Take actions necessary to decommission the plant and continue to maintain the facility, including, where applicable, the storage, control and maintenance of the spent fuel, in a safe condition; and

B. Conduct activities in accordance with all other restrictions applicable to the facility in accordance with the NRC regulations and the applicable provisions of the 10 CFR 50 facility license as defined in Section 2 of this license.

FOR THE NUCLEAR REGULATORY COMMISSION

Original Signed by
Harold R. Denton

Harold R. Denton, Director
Office of Nuclear Reactor Regulation

Attachments:
1. Attachment 1 - Deleted by Amendment No. 176
2. Appendix A (Technical Specifications)
3. Appendix B (Environmental Protection Plan)
4. Appendix C (Antitrust Conditions)

Date of Issuance: NOV 15 1982

APPENDIX A

TO THE

FACILITY OPERATING LICENSE NPF-10

AND

FACILITY OPERATING LICENSE NPF-15

TECHNICAL SPECIFICATIONS FOR

SAN ONOFRE NUCLEAR GENERATING STATION UNIT 2 AND UNIT 3

TABLE OF CONTENTS

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1.0 USE AND APPLICATION

1.1 Definitions

---------------------------------NOTE---------------------------------
The defined terms of this section appear in capitalized type and are applicable throughout these Technical Specifications and Bases.

Term	Definition
ACTIONS	ACTIONS shall be that part of a Specification that prescribes Required Actions to be taken under designated Conditions within specified Completion Times.
CERTIFIED FUEL HANDLER	A CERTIFIED FUEL HANDLER is an individual who complies with provisions of the CERTIFIED FUEL HANDLER training program required by TS 5.3.2.
OPERABLE–OPERABILITY	A system, subsystem, train, component, or device shall be OPERABLE when it is capable of performing its specified safety function(s) and when all necessary attendant instrumentation, controls, normal or emergency electrical power, cooling and seal water, lubrication, and other auxiliary equipment that are required for the system, subsystem, train, component, or device to perform its specified safety function(s) are also capable of performing their related support function(s).

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1.0 USE AND APPLICATION

1.2 Logical Connectors

PURPOSE

The purpose of this section is to explain the meaning of logical connectors.

Logical connectors are used in Technical Specifications (TS) to discriminate between, and yet connect, discrete Conditions, Required Actions, Completion Times, Surveillances, and Frequencies. The only logical connectors that appear in TS are AND and OR. The physical arrangement of these connectors constitutes logical conventions with specific meanings.

BACKGROUND

Several levels of logic may be used to state Required Actions. These levels are identified by the placement (or nesting) of the logical connectors and by the number assigned to each Required Action. The first level of logic is identified by the first digit of the number assigned to a Required Action and the placement of the logical connector in the first level of nesting (i.e., left justified with the number of the Required Action). The successive levels of logic are identified by additional digits of the Required Action number and by successive indentions of the logical connectors.

When logical connectors are used to state a Condition, Completion Time, Surveillance, or Frequency, only the first level of logic is used, and the logical connector is left justified with the statement of the Condition, Completion Time, Surveillance, or Frequency.

EXAMPLE

The following example illustrates the use of logical connectors.

EXAMPLE 1.2-1

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. LCO not met.	A.1 Verify . . . AND A.2 Restore . . .	

In this example the logical connector AND is used to indicate that when in Condition A, both Required Actions A.1 and A.2 must be completed.

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1.0 USE AND APPLICATION

1.3 Completion Times

PURPOSE	The purpose of this section is to establish the Completion Time convention and to provide guidance for its use.
BACKGROUND	Limiting Condition for Operation (LCOs) specify minimum requirements for ensuring safe storage of fuel assemblies. The ACTIONS associated with an LCO state Conditions that typically describe the ways in which the requirements of the LCO can fail to be met. Specified with each stated Condition are Required Action(s) and Completion Time(s).
DESCRIPTION	The Completion Time is the amount of time allowed for completing a Required Action. It is referenced to the time of discovery of a situation (e.g., inoperable equipment or variable not within limits) that requires entering an ACTIONS Condition unless otherwise specified, providing the facility is in a specified condition stated in the Applicability of the LCO. Required Actions must be completed prior to the expiration of the specified Completion Time. An ACTIONS Condition remains in effect and the Required Actions apply until the Condition no longer exists or the facility is not within the LCO Applicability.

EXAMPLE

The following example illustrates the use of Completion Times.

EXAMPLE 1.3-1

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Required Action and associated Completion Time not met.	B.1 Verify . . . AND B.2 Restore . . .	6 hours 36 hours

Condition B has two Required Actions. Each Required Action has its own separate Completion Time. Each Completion Time is referenced to the time that Condition B is entered.

The Required Actions of Condition B are to perform the verification within 6 hours AND perform the restoration within 36 hours. A total of 6 hours is allowed for performing the verification and a total of 36 hours (not 42 hours) is allowed for performing the restoration from the time that Condition B was entered. If verification is performed within 3 hours, the

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1.3 Completion Times

EXAMPLE (continued)	
	time allowed for performing the restoration is the next 33 hours because the total time allowed for performing the restoration is 36 hours.
IMMEDIATE COMPLETION TIME	When "Immediately" is used as a Completion Time, the Required Action should be pursued without delay and in a controlled manner.

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1.0 USE AND APPLICATION

1.4 Frequency

PURPOSE	The purpose of this section is to define the proper use and application of Frequency requirements.
DESCRIPTION	Each Surveillance Requirement (SR) has a specified Frequency in which the Surveillance must be met in order to meet the associated LCO. An understanding of the correct application of the specified Frequency is necessary for compliance with the SR. The "specified Frequency" is referred to throughout this section and each of the Specifications of Section 3.0, Surveillance Requirement (SR) Applicability. The "specified Frequency" consists of the requirements of the Frequency column of each SR, as well as certain Notes in the Surveillance column that modify performance requirements.
EXAMPLES	The following examples illustrate the various ways that Frequencies are specified. In these examples, the Applicability of the LCO (LCO not shown) occurs whenever any fuel assembly is stored in the fuel storage pool.

EXAMPLE 1.4-1

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
Verify . . .	7 days

Example 1.4-1 contains the type of SR most often encountered in the Technical Specifications (TS). The Frequency specifies an interval (7 days) during which the associated Surveillance must be performed at least one time. Performance of the Surveillance initiates the subsequent interval. Although the Frequency is stated as 7 days, an extension of the time interval to 1.25 times the stated Frequency is allowed by SR 3.0.2 for operational flexibility. The measurement of this interval continues at all times, even when the SR is not required to be met per SR 3.0.1 (such as when the equipment is inoperable, a variable is outside specified limits, or the facility is outside the Applicability of the LCO). If the interval specified by SR 3.0.2 is exceeded while the facility is in a specified condition in the Applicability of the LCO, then SR 3.0.3 becomes applicable.

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1.4 Frequency

EXAMPLES (continued)

If the interval as specified by SR 3.0.2 is exceeded while the facility is not in a specified condition in the Applicability of the LCO for which performance of the SR is required, the Surveillance must be performed within the Frequency requirements of SR 3.0.2 prior to entry into the specified condition. Failure to do so would result in a violation of SR 3.0.4.

EXAMPLE 1.4-2

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
Verify . . .	Prior to moving a fuel assembly . . .

Example 1.4-2 illustrates a one time performance Frequency.

This type of Frequency does not qualify for the 25% extension allowed by SR 3.0.2.

3.0 LIMITING CONDITION FOR OPERATION (LCO) APPLICABILITY

LCO 3.0.1	LCOs shall be met during the specified conditions in the Applicability, except as provided in LCO 3.0.2.
LCO 3.0.2	Upon discovery of a failure to meet an LCO, the Required Actions of the associated Conditions shall be met. If the LCO is met or is no longer applicable prior to expiration of the specified Completion Time(s), completion of the Required Action(s) is not required, unless otherwise stated.

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3.0 SURVEILLANCE REQUIREMENT (SR) APPLICABILITY

SR 3.0.1	SRs shall be met during the specified conditions in the Applicability for individual LCOs, unless otherwise stated in the SR. Failure to meet a Surveillance, whether such failure is experienced during the performance of the Surveillance or between performances of the Surveillance, shall be failure to meet the LCO. Failure to perform a Surveillance within the specified Frequency shall be failure to meet the LCO except as provided in SR 3.0.3. Surveillances do not have to be performed on inoperable equipment or variables outside specified limits.
SR 3.0.2	The specified Frequency for each SR is met if the Surveillance is performed within 1.25 times the interval specified in the Frequency, as measured from the previous performance or as measured from the time a specified condition of the Frequency is met.
SR 3.0.3	If it is discovered that a Surveillance was not performed within its specified Frequency, then compliance with the requirement to declare the LCO not met may be delayed, from the time of discovery, up to 24 hours or up to the limit of the specified Frequency, whichever is greater. This delay period is permitted to allow performance of the Surveillance. A risk evaluation shall be performed for any Surveillance delayed greater than 24 hours and the risk impact shall be managed. If the Surveillance is not performed within the delay period, the LCO must immediately be declared not met, and the applicable Condition(s) must be entered. The Completion Times of the Required Actions begin immediately upon expiration of the delay period. When the Surveillance is performed within the delay period and the Surveillance is not met, the LCO must immediately be declared not met, and the applicable Condition(s) must be entered. The Completion Times of the Required Actions begin immediately upon failure to meet the Surveillance.
SR 3.0.4	Entry into a specified condition in the Applicability of an LCO shall not be made unless the LCO's Surveillances have been met within their specified Frequency. This provision shall not prevent entry into specified conditions in the Applicability that are required to comply with ACTIONS.

3.1 PLANT SYSTEMS

3.1.1 Fuel Storage Pool Water Level

LCO 3.1.1 The fuel storage pool water level shall be ≥ 23 ft over the top of irradiated fuel assemblies seated in the storage racks.

APPLICABILITY: During movement of fuel assemblies in the fuel storage pool.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. Fuel storage pool water level not within limit.	A.1	Suspend movement of fuel assemblies in fuel storage pool.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.1.1	Verify the fuel storage pool water level is ≥ 23 ft above the top of irradiated fuel assemblies seated in the storage racks.	7 days

3.1 PLANT SYSTEMS

3.1.2 Fuel Storage Pool Boron Concentration

LCO 3.1.2 The fuel storage pool boron concentration shall be ≥ 2000 ppm.

APPLICABILITY: Whenever any fuel assembly is stored in the fuel storage pool.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Fuel storage pool boron concentration not within limit.	A.1 Suspend movement of fuel assemblies in the fuel storage pool.	Immediately
	<u>AND</u>	
	A.2 Initiate action to restore fuel storage pool boron concentration to within limit.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.2.1	Verify the fuel storage pool boron concentration is within limit.	7 days

3.1 PLANT SYSTEMS

3.1.3 Spent Fuel Assembly Storage

LCO 3.1.3

The combination of initial enrichment and burnup of each SONGS 2 and 3 spent fuel assembly stored in Region I shall be within the acceptable burnup domain of Figure 3.1.3-1 or Figure 3.1.3-2 or the fuel assembly shall be stored in accordance with Technical Specification 4.3.1.1.

The combination of initial enrichment and burnup of each SONGS 2 and 3 spent fuel assembly stored in Region II shall be within the acceptable burnup domain of Figure 3.1.3-3 or Figure 3.1.3-4, or the fuel assembly shall be stored in accordance with Technical Specification 4.3.1.1.

Each SONGS 1 uranium dioxide spent fuel assembly stored in Region II shall be stored in accordance with Technical Specification 4.3.1.1.

APPLICABILITY: Whenever any fuel assembly is stored in the fuel storage pool.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Requirements of the LCO not met.	A.1 Initiate action to bring the noncomplying fuel assembly into compliance.	Immediately

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.1.3.1	Verify by administrative means the initial enrichment, burnup, and cooling time of the fuel assembly are in accordance with LCO 3.1.3, or Design Features 4.3.1.1, or Licensee Controlled Specification (LCS) 4.0.100. Rev 2, dated 09/27/07.	Prior to moving a fuel assembly to any spent fuel pool storage location.

FIGURE 3.1.3-1

MINIMUM BURNUP AND COOLING TIME VS. INITIAL ENRICHMENT
FOR
UNRESTRICTED PLACEMENT OF SONGS 2 AND 3 FUEL
IN
REGION I RACKS

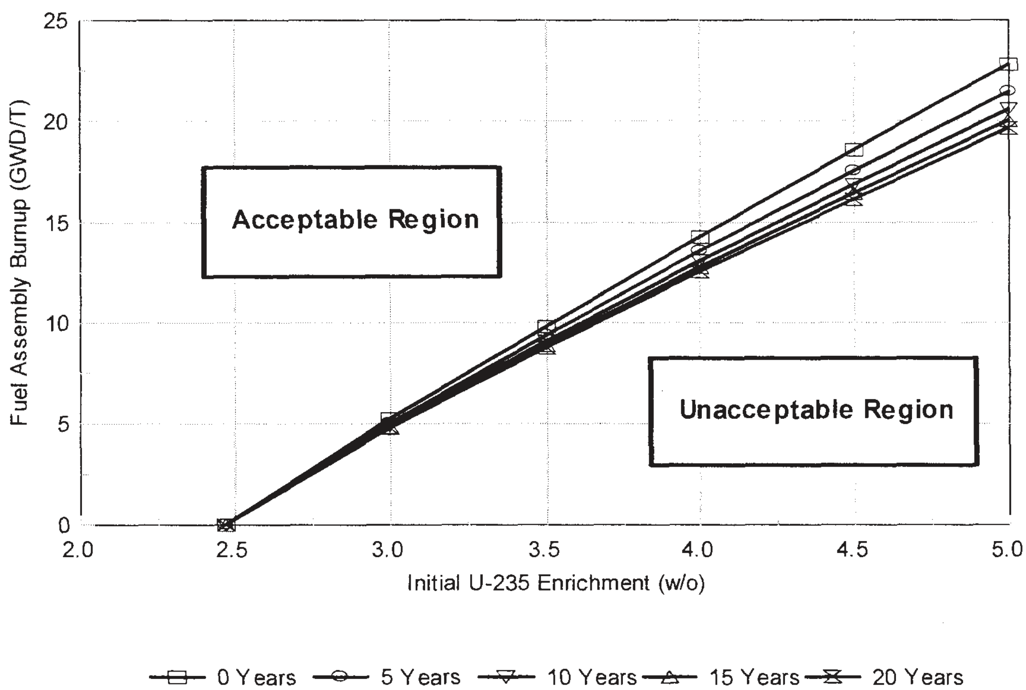

FIGURE 3.1.3-2

MINIMUM BURNUP AND COOLING TIME VS. INITIAL ENRICHMENT
FOR
PLACEMENT OF SONGS 2 AND 3 FUEL IN PERIPHERAL POOL LOCATIONS
IN
REGION I RACKS

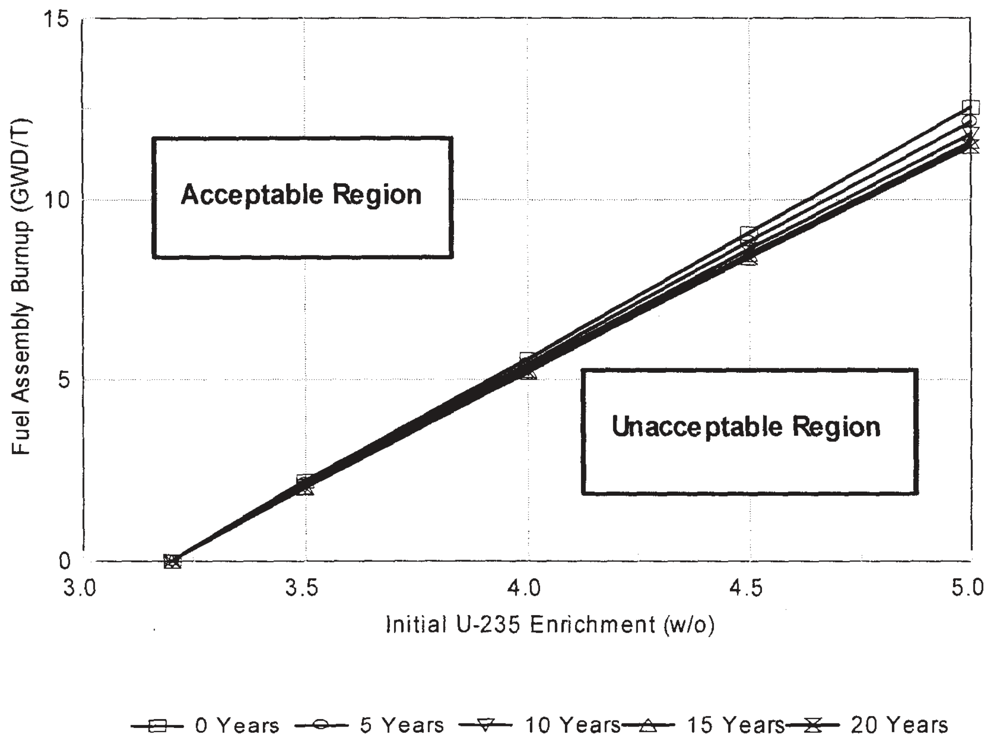

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FIGURE 3.1.3-3

MINIMUM BURNUP AND COOLING TIME VS. INITIAL ENRICHMENT
FOR
UNRESTRICTED PLACEMENT OF SONGS 2 AND 3 FUEL
IN
REGION II RACKS

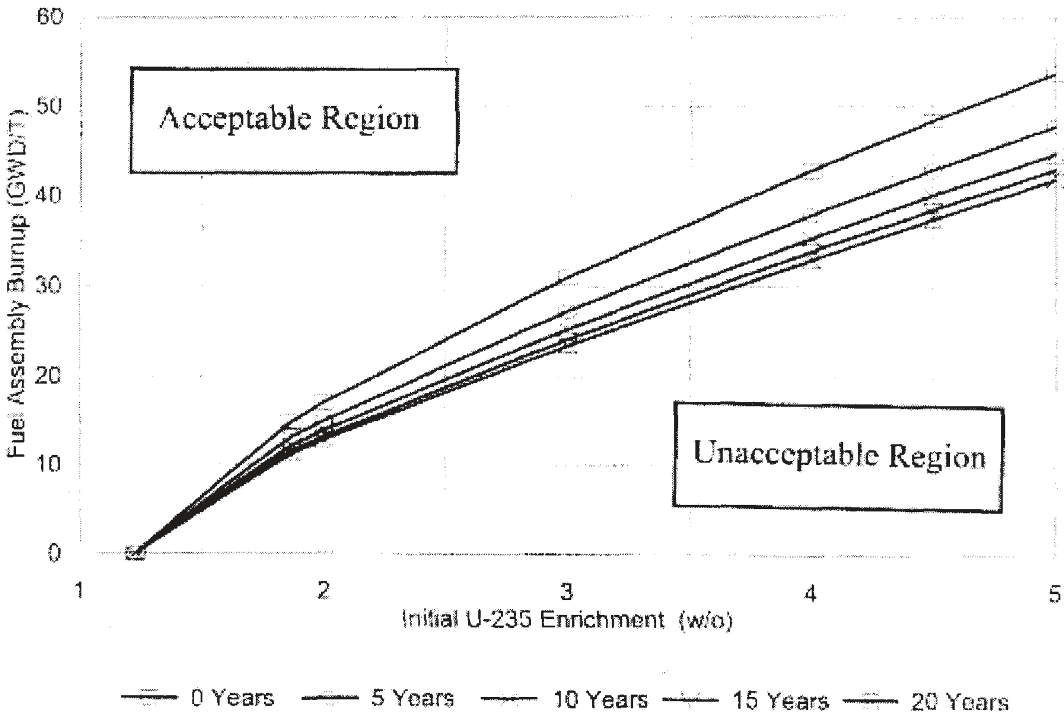
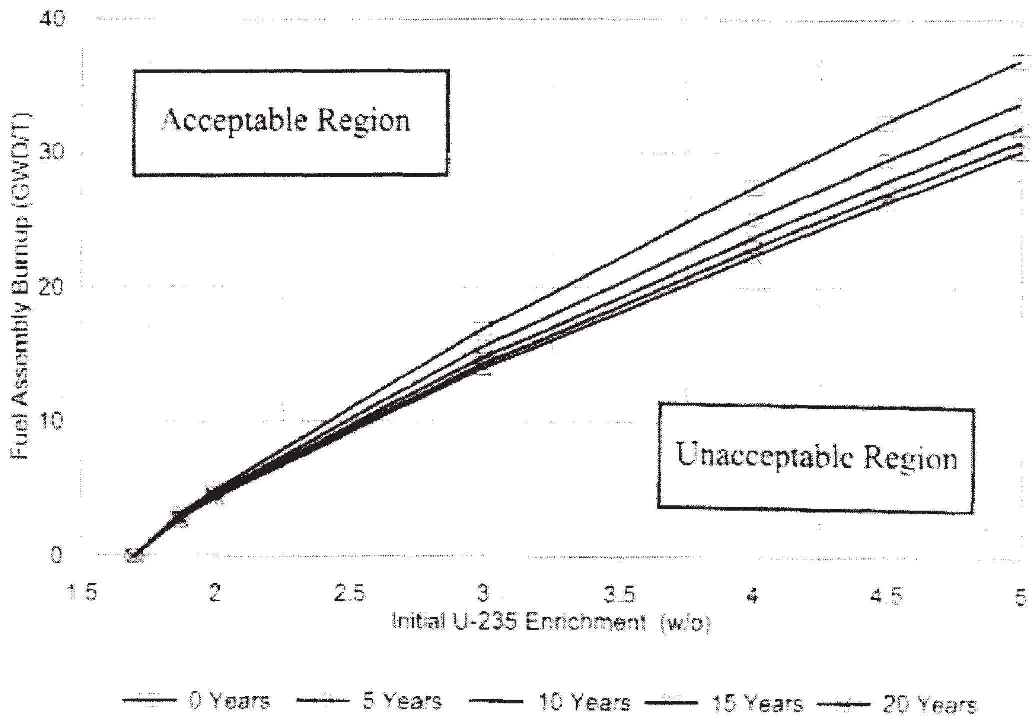


FIGURE 3.1.3-4

MINIMUM BURNUP AND COOLING TIME VS. INITIAL ENRICHMENT
FOR
PLACEMENT OF SONGS 2 AND 3 FUEL IN PERIPHERAL POOL LOCATIONS
IN
REGION II RACKS



4.0 DESIGN FEATURES

4.1 Site

4.1.1 Exclusion Area Boundary

The exclusion area boundary shall be as shown in Figure 4.1-1.

4.1.2 Low Population Zone (LPZ)

The LPZ shall be as shown in Figure 4.1-2.

4.2 Deleted.

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4.0 DESIGN FEATURES (continued)

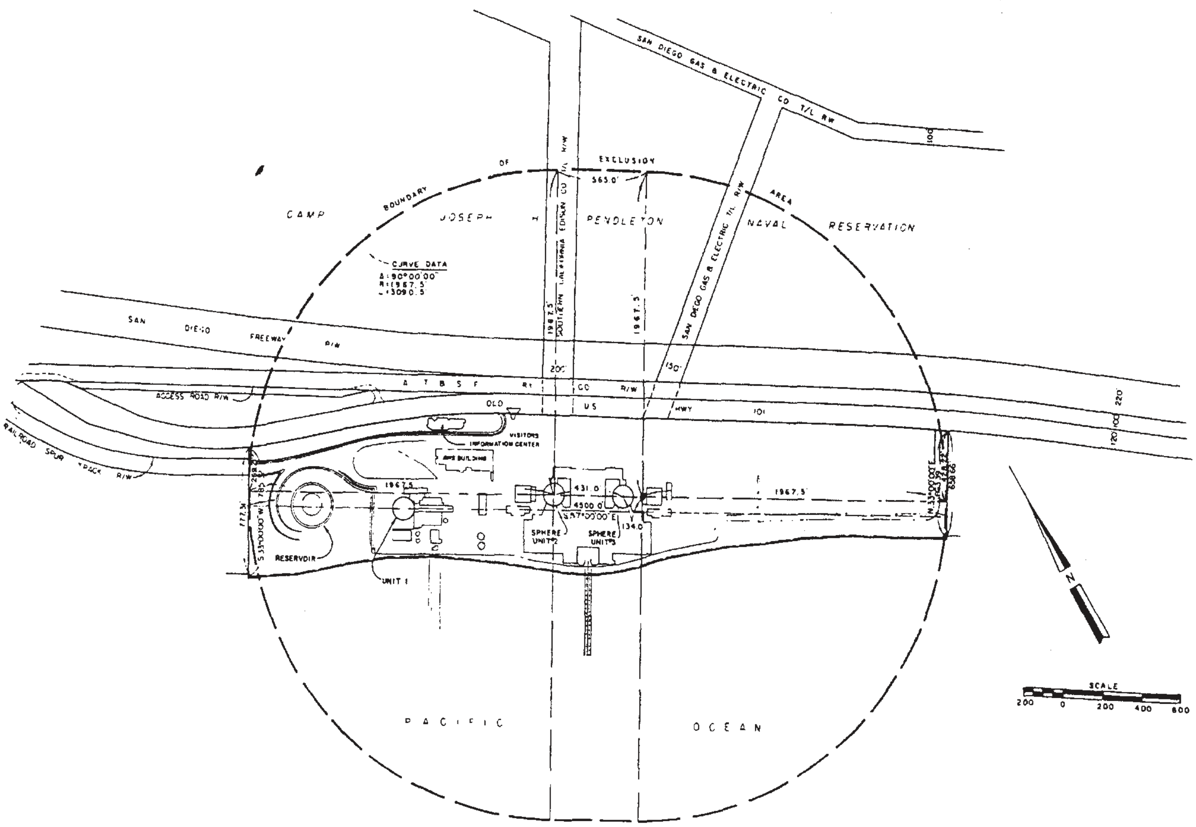

Figure 4.1-1 (page 1 of 1)
Exclusion Area Boundary

4.0 DESIGN FEATURES (continued)

Figure 4.1-2 (page 1 of 1)
Low Population Zone

4.0 DESIGN FEATURES (continued)

4.3 Fuel Storage

4.3.1 Criticality

4.3.1.1 The spent fuel storage racks are designed and shall be maintained with:

a. Fuel assemblies having a maximum U-235 enrichment of 4.8 weight percent;

b. $K_{eff} < 1.0$ if fully flooded with unborated water, which includes an allowance for uncertainties as described in Section 9.1 of the UFSAR;

c. $K_{eff} \leq 0.95$ if fully flooded with water borated to 1700 ppm, which includes an allowance for uncertainties as described in Section 9.1 of the UFSAR;

d. Three or five Borated stainless steel guide tube inserts (GT-Insert) may be used. When three borated stainless steel guide tube inserts are used, they will be installed in an assembly's center guide tube, the guide tube associated with the serial number, and the diagonally opposite guide tube. Fuel containing GT-Inserts may be placed in either Region I or Region II. However, credit for GT-Inserts is only taken for Region II storage.

A five-finger CEA may be installed in an assembly. Fuel containing a five-finger CEA may be placed in either Region I or Region II. Credit for inserted 5-finger CEAs is taken for both Region I and Region II.

e. A nominal 8.85 inch center to center distance between fuel assemblies placed in Region II;

f. A nominal 10.40 inch center to center distance between fuel assemblies placed in Region I;

g. Prior to using the storage criteria of LCO 3.1.3 and LCS 4.0.100, the following uncertainties will be applied:

(1) The calculated discharge burnup of San Onofre Units 2 and 3 assemblies will be reduced by 6.6%.

(2) The calculated discharge burnup of San Onofre Unit 1 fuel assemblies will be reduced by 10.0%.

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4.0 DESIGN FEATURES

4.3 Fuel Storage (continued)

h. Units 2 and 3 fuel assemblies with a burnup in the "acceptable range" of Figure 3.1.3-1 are allowed unrestricted storage in Region I;

i. Units 2 and 3 fuel assemblies with a burnup in the "acceptable range" of Figure 3.1.3-2 are allowed unrestricted storage in the peripheral pool locations with 1 or 2 faces toward the spent fuel pool walls of Region I;

j. Units 2 and 3 fuel assemblies with a burnup in the "acceptable range" of Figure 3.1.3-3 are allowed unrestricted storage in Region II;

k. Units 2 and 3 fuel assemblies with a burnup in the "acceptable range" of Figure 3.1.3-4 are allowed unrestricted storage in the peripheral pool locations with 1 or 2 faces toward the spent fuel pool walls of Region II;

l. Units 2 and 3 fuel assemblies with a burnup in the "unacceptable range" of Figure 3.1.3-1, Figure 3.1.3-2, Figure 3.1.3-3, and Figure 3.1.3-4 will be stored in compliance with Licensee Controlled Specification 4.0.100 Rev. 2, dated 9/27/07; and

m. Each SONGS 1 uranium dioxide spent fuel assembly stored in Region II shall be stored in accordance with Licensee Controlled Specification 4.0.100 Rev. 2, dated 9/27/07.

4.3.2 Drainage

The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below Technical Specification 3.1.1 value (23 feet above the top of irradiated fuel assemblies seated in the storage racks).

4.3.3 Capacity

The spent fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than 1542 fuel assemblies.

5.0 ADMINISTRATIVE CONTROLS

5.1 Responsibility

5.1.1 The corporate officer with direct responsibility for the plant shall be responsible for overall management of the San Onofre Nuclear Generating Station, and all site support functions. He shall delegate in writing the succession to this responsibility during his absence.

5.1.2 The Shift Manager shall be responsible for the ultimate command decision authority for all unit activities which affect the safety of the plant, site personnel, and/or the general public.

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5.0 ADMINISTRATIVE CONTROLS

5.2 Organization

5.2.1 Onsite and Offsite Organizations

Onsite and offsite organizations shall be established for plant operation and corporate management, respectively. The onsite and offsite organizations shall include the positions for activities affecting the safety of the nuclear fuel.

a. Lines of authority, responsibility, and communication shall be established and defined for the highest management levels through intermediate levels to and including all operating organization positions. These relationships shall be documented and updated, as appropriate, in the form of organization charts, functional descriptions of departmental responsibilities and relationships, and job descriptions for key personnel positions, or in equivalent forms of documentation. These relationships, including the plant-specific titles of those personnel fulfilling the responsibilities for the positions delineated in these Technical Specifications, are documented in the UFSAR.

b. The corporate officer with direct responsibility for the plant shall be responsible for overall safe handling and storage of nuclear fuel and shall have control over those onsite activities necessary for safe handling and storage of the nuclear fuel.

c. A specified corporate officer (or officers) shall have corporate responsibility for overall plant nuclear safety and shall take any measures needed to ensure acceptable performance of the staff in operating, maintaining, and providing technical support to the plant to ensure safe management of nuclear fuel.

d. The individuals who train CERTIFIED FUEL HANDLERS, and those who carry out radiation protection and quality assurance functions may report to the appropriate onsite manager; however, they shall have sufficient organizational freedom to ensure their ability to perform their assigned functions.

5.2.2 FACILITY STAFF

The facility staff organization shall include the following:

a. Each on duty shift shall be composed of at least the minimum shift crew composition shown in Table 5.2.2-1.

b. At least one person qualified as Emergency Coordinator/Emergency Director shall be in the Control Room when nuclear fuel is stored in the spent fuel pools.

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5.2 Organization

5.2.2 FACILITY STAFF (continued)

c. Shift crew composition may be less than the minimum requirement of Table 5.2.2-1 for a period of time not to exceed 2 hours in order to accommodate unexpected absence of on-duty shift crew members provided immediate action is taken to restore the shift crew composition to within the minimum requirements. During such absences, no fuel movement or movement of heavy loads over storage racks containing fuel is permitted.

d. Oversight of fuel handling operations shall be provided by a CERTIFIED FUEL HANDLER.

e. The Shift Manager shall be a CERTIFIED FUEL HANDLER.

f. An individual qualified in radiation protection procedures shall be on site during fuel handling operations or movement of loads over the storage racks containing fuel.

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5.2 Organization (continued)

Table 5.2.2-1
Minimum Shift Crew Composition

POSITION	MINIMUM STAFFING
CERTIFIED FUEL HANDLER	1*
Certified Operator	1

Note: The Certified Operator position may be filled by a CERTIFIED FUEL HANDLER.
* May be shared between Units 2 and 3.

5.0 ADMINISTRATIVE CONTROLS

5.3 Facility Staff Qualifications

5.3.1 Each member of the facility staff shall meet or exceed the minimum qualifications of ANSI N18.1-1971 for comparable positions, except: a) the radiation protection manager who shall meet or exceed the qualifications of Regulatory Guide 1.8, September 1975.

5.3.2 An NRC approved training and retraining program for the CERTIFIED FUEL HANDLERS shall be maintained.

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5.0 ADMINISTRATIVE CONTROLS

5.4 Technical Specifications (TS) Bases Control

5.4.1 Changes to the Bases of the TS shall be made under appropriate administrative controls.

5.4.2 Changes to the Bases may be made without prior NRC approval provided the changes do not require either of the following:

a. A change in the TS incorporated in the license; or

b. A change to the updated UFSAR or Bases that requires NRC approval pursuant to 10 CFR 50.59.

5.4.3 The Bases Control Program shall contain provisions to ensure that the Bases are maintained consistent with the UFSAR.

5.4.4 Proposed changes that meet the criteria of (a) or (b) above shall be reviewed and approved by the NRC prior to implementation. Changes to the Bases implemented without prior NRC approval shall be provided to the NRC every 24 months.

5.0 ADMINISTRATIVE CONTROLS

5.5 Procedures, Programs, and Manuals

5.5.1 Procedures

5.5.1.1 Scope

Written procedures shall be established, implemented, and maintained covering the following activities:

a. The applicable procedures recommended in Regulatory guide 1.33, Revision 2, Appendix A, February 1978;

b. Deleted.

c. Quality assurance for effluent and environmental monitoring using the guidance in Regulatory Guide 4.15, Revision 1, 1979;

d. Fire Protection Program implementation; and

e. Programs, as specified in Specification 5.5.2.

5.5.2 Programs and Manuals

The following programs and manuals shall be established, implemented, and maintained.

5.5.2.1 Offsite Dose Calculation Manual (ODCM)

a. The ODCM shall contain the methodology and parameters used in the calculation of offsite doses resulting from radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring alarm and trip setpoints, and in the conduct of the Radiological Environmental Monitoring Program;

b. The ODCM shall also contain the Radioactive Effluent Controls required by Specification 5.5.2.3 and Radiological Environmental Monitoring programs required by the LCS, and descriptions of the information that should be included in the Annual Radiological Environmental Operating Report and the Radioactive Effluent Release Report required by Specification 5.7.1.2 and Specification 5.7.1.3.

5.5.2.1.1 Licensee-initiated changes to the ODCM:

a. Shall be documented and records of reviews performed shall be retained. This documentation shall contain:

5.5 Procedures, Programs, and Manuals

5.5.2.1.1 Licensee-initiated changes to the ODCM (continued):

1. Sufficient information to support the change(s) together with the appropriate analyses or evaluations justifying the change(s);

2. A determination that the change(s) maintain the levels of radioactive effluent control required by 10 CFR 20.106, 40 CFR 190, 10 CFR 50.36a, and 10 CFR 50, Appendix I, and not adversely impact the accuracy or reliability of effluent, dose, or setpoint calculations.

3. Documentation of the fact that the change has been reviewed and found acceptable.

b. Shall become effective upon review and approval by the corporate officer with direct responsibility for the plant or designee.

c. Shall be submitted to the NRC in the form of a complete, legible copy of the entire ODCM as a part of or concurrent with the Radioactive Effluent Release Report for the period of the report in which any change in the ODCM was made. Each change shall be identified by markings in the margin of the affected pages, clearly indicating the area of the page that was changed, and shall indicate the date (i.e., month and year) the change was implemented.

5.5.2.2 Deleted

5.5.2.3 Radioactive Effluent Controls Program

This program conforming to 10 CFR 50.36a provides for the control of radioactive effluents and for maintaining the doses to members of the public from radioactive effluents as low as reasonably achievable. The program shall be contained in the ODCM, shall be implemented by operating procedures, and shall include remedial actions to be taken whenever the program limits are exceeded. The program shall include the following elements:

a. Limitations on the functional capability of radioactive liquid and gaseous monitoring instrumentation including surveillance tests and setpoint determination in accordance with the methodology in the ODCM;

b. Limitations on the concentrations of radioactive material released in liquid effluents to unrestricted areas, conforming to 10 CFR 20, Appendix B, Table II, Column 2;

c. Monitoring, sampling, and analysis of radioactive liquid and gaseous effluents in accordance with 10 CFR 20.106 and with the methodology and parameters in the ODCM;

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5.5 Procedures, Programs, and Manuals

5.5.2.3 Radioactive Effluent Controls Program (continued)

d. Limitations on the annual and quarterly doses or dose commitment to a member of the public from radioactive materials in liquid effluents released from each unit to unrestricted areas, conforming to 10 CFR 50, Appendix I;

e. Determination of cumulative and projected dose contributions from radioactive effluents for the current calendar quarter and current calendar year in accordance with the methodology and parameters in the ODCM at least every 31 days;

f. Limitations on the functional capability and use of the liquid and gaseous effluent treatment systems to ensure that appropriate portions of these systems are used to reduce releases of radioactivity when the projected doses in a period of 31 days would exceed 2 percent of the guidelines for the annual dose or dose commitment, conforming to 10 CFR 50, Appendix I;

g. Limitations on the dose rate resulting from radioactive material released in gaseous effluents to areas beyond the site boundary conforming to the dose associated with 10 CFR 20, Appendix B, Table II, Column 1;

h. Limitations on the annual and quarterly air doses resulting from noble gases released in gaseous effluents from each unit to areas beyond the site boundary, conforming to 10 CFR 50, Appendix I;

i. Limitations on the annual and quarterly doses to a member of the public from iodine-131, iodine-133, tritium, and all radionuclides in particulate form with half lives greater than 8 days in gaseous effluents released from each unit to areas beyond the site boundary, conforming to 10 CFR 50, Appendix I; and

j. Limitations on the annual dose or dose commitment to any member of the public due to releases of radioactivity and to radiation from uranium fuel cycle sources, conforming to 40 CFR 190.

5.5.2.4 Deleted

5.5.2.5 Deleted

5.5.2.6 Deleted

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5.5 Procedures, Programs, and Manuals (continued)

5.5.2.7 Storage Tank Radioactivity Monitoring Program

This program provides controls for the quantity of radioactivity contained in unprotected outdoor liquid storage tanks. The liquid radwaste quantities shall be determined in accordance with Standard Review Plan, Section 15.7.3, "Postulated Radioactive Release due to Tank Failures".

The program shall include a surveillance program to ensure that the quantity of radioactivity contained in all outdoor liquid radwaste tanks that are not surrounded by liners, dikes, or walls, capable of holding the tanks' contents and that do not have tank overflows and surrounding area drains connected to the Liquid Waste Management System is less than the amount that would result in concentrations less than the limits of 10 CFR Part 20, Appendix B, Table II, Column 2, at the nearest potable water supply and the nearest surface water supply in an unrestricted area, in the event of an uncontrolled release of the tanks' contents.

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Storage Tank Radioactivity Monitoring Program surveillance frequencies.

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5.0 ADMINISTRATIVE CONTROLS

5.6 Deleted

5.0 ADMINISTRATIVE CONTROLS

5.7 Reporting Requirements

5.7.1 Routine Reports

In addition to the applicable reporting requirements of Title 10, Code of Federal Regulations, the following reports shall be submitted in accordance with 10 CFR 50.4. The reports shall be addressed to the U.S. Nuclear Regulatory Commission, Attention: Document Control Desk, Washington, D.C., with a copy to the Regional Administrator of the Regional Office of the NRC, unless otherwise noted.

5.7.1.1 Deleted

5.7.1.2 Annual Radiological Environmental Operating Report

The Annual Radiological Environmental Operating Report covering the operation of the facility during the previous calendar year shall be submitted by May 15 of each year. The report shall include summaries, interpretations, and analyses of trends of the results of the Radiological Environmental Monitoring Program for the reporting period. The material provided shall be consistent with the objectives outlined in the Offsite Dose Calculation Manual (ODCM), and in 10 CFR 50, Appendix I, Sections IV.B.2, IV.B.3, and IV.C.

The Annual Radiological Environmental Operating Report shall include the results of analyses of all radiological environmental samples and of all environmental radiation measurements taken during the period pursuant to the locations specified in the table and figures in the ODCM, as well as summarized and tabulated results of these analyses and measurements in the format of the table in the Radiological Assessment Branch Technical Position, Revision 1, November 1979. The report shall identify the thermoluminescent dosimeter (TLD) results that represent collocated dosimeters in relation to the NRC TLD program and the exposure period associated with each result. In the event that some individual results are not available for inclusion with the report, the report shall be submitted noting and explaining the reasons for the missing results. The missing data shall be submitted in a supplementary report as soon as possible.

5.7 Reporting Requirements (continued)

5.7.1.3 Radiological Effluent Release Report

The Radioactive Effluent Release Report covering the operation of the facility during the previous calendar year shall be submitted before May 1 of each year. The report shall include a summary of the quantities of radioactive liquid and gaseous effluents released from the facility. The report shall also include a summary of the quantities of solid radioactive waste shipped from the facility directly to the disposal site and quantities of solid radioactive waste shipped from the facility's intermediary processor to the disposal site. The material provided shall be consistent with the objectives outlined in the ODCM and Process Control Program (PCP) and in conformance with 10 CFR 50.36a and 10 CFR 50, Appendix I, Section IV.B.1.

5.0 ADMINISTRATIVE CONTROLS

5.8 High Radiation Area

5.8.1 Each high radiation area as defined 10 CFR 20 shall be barricaded and conspicuously posted as a high radiation area, and entrance thereto shall be controlled by requiring issuance of a Radiation Exposure Permit (REP).

Any individual or group of individuals permitted to enter such areas shall be provided with or accompanied by one or more of the following:

a. A radiation monitoring device that continuously indicates the radiation dose rate in the area,

b. A radiation monitoring device that continuously integrates the radiation dose rate in the area and alarms when a preset integrated dose is received. Entry into such areas with this monitoring device may be made after the dose rates in the area have been determined and personnel have been made knowledgeable of them,

c. An individual qualified in radiation protection procedures with a radiation dose rate monitoring device. This individual is responsible for providing positive radiation protection control over the activities within the area and shall perform periodic radiation surveillance at the frequency specified in the radiation protection procedures or the applicable REP.

5.8.2 In addition, areas that are accessible to personnel and that have radiation levels greater than 1.0 rem (but less than 500 rads at 1 meter) in 1 hour at 30 cm from the radiation source, or from any surface penetrated by the radiation, shall be provided with locked doors to prevent unauthorized entry, and the keys shall be maintained under the administrative control of the shift manager on duty or radiation protection supervisor. Doors shall remain locked except during periods of access by personnel under an approved REP that specifies the dose rates in the immediate work areas and the maximum allowable stay time for individuals in that area. In lieu of a stay time specification on the REP, direct or remote continuous surveillance (such as closed circuit TV cameras) may be made by personnel qualified in radiation protection procedures to provide positive exposure control over the activities being performed within the area.

5.8.3 Individual high radiation areas that are accessible to personnel, that could result in radiation doses greater than 1.0 rem in 1 hour, and that are within large areas where no enclosure exists to enable locking and where no enclosure can be reasonably constructed around the individual area shall be barricaded and conspicuously posted. A flashing light shall be activated as a warning device whenever the dose rate in such an area exceeds or is expected to exceed 1.0 rem in 1 hour at 30 cm from the radiation source or from any surface penetrated by the radiation.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 230 TO FACILITY OPERATING LICENSE NO. NPF-10

AND AMENDMENT NO. 223 TO FACILITY OPERATING LICENSE NO. NPF-15

SOUTHERN CALIFORNIA EDISON COMPANY

SAN DIEGO GAS AND ELECTRIC COMPANY

THE CITY OF RIVERSIDE, CALIFORNIA

SAN ONOFRE NUCLEAR GENERATING STATION, UNITS 2 AND 3

DOCKET NOS. 50-361 AND 50-362

1.0 INTRODUCTION

By letter dated June 12, 2013 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML131640201), Southern California Edison (SCE, the licensee) submitted a certification to the U.S. Nuclear Regulatory Commission (NRC) indicating its intention to permanently cease power operations at San Onofre Nuclear Generating Station (SONGS), Units 2 and 3 as of June 7, 2013, pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) paragraph 50.82(a)(1)(i). By letters dated June 28, 2013 (ADAMS Accession No. ML13183A391), and July 22, 2013 (ADAMS Accession No. ML13204A304), SCE submitted certifications of permanent removal of fuel from the Unit 3 and Unit 2 reactor vessels as of October 5, 2012, and July 18, 2013, respectively, pursuant to 10 CFR 50.82(a)(1)(ii). Upon docketing of these certifications, and pursuant to 10 CFR 50.82(a)(2), the SONGS Units 2 and 3 facility operating licenses no longer authorize operation of the reactors or emplacement or retention of fuel into the reactor vessels. Spent fuel is currently stored onsite in the spent fuel pools (SFPs) and in the onsite independent spent fuel storage installation (ISFSI).

By letter dated March 21, 2014 (ADAMS Accession No. ML14085A141), as supplemented by letters dated October 1, 2014; and February 23, February 25, and March 18, 2015 (ADAMS Accession Nos. ML14280A264, ML15058A030, ML15058A033, and ML15082A017, respectively), SCE submitted a license amendment request consisting of amendment applications to Facility Operating License Nos. NPF-10 and NPF-15 for SONGS Units 2 and 3, respectively. The proposed amendments would revise the facility operating licenses and revise the associated technical specifications (TSs) to reflect the permanent cessation of operations of SONGS Units 2 and 3.

The supplemental letters dated October 1, 2014; and February 23, February 25, and March 18, 2015, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the NRC staff's original proposed no

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significant hazards consideration determination as published in the *Federal Register* on September 16, 2014 (79 FR 55513).

As stated above, pursuant to 10 CFR 50.82(a)(2), the 10 CFR Part 50 licenses for SONGS Units 2 and 3 no longer authorize operation of the reactors or emplacement or retention of fuel in the reactor vessels. SONGS Units 2 and 3 have been shut down since January 2012. At the time of the licensee's submittal, the fission product inventory of all spent fuel that is stored in the SONGS Units 2 and 3 spent fuel pools had decayed more than two years since last irradiated in the reactor core. SONGS Unit 1 was permanently shut down in 1993 and is already in the decommissioning phase where above ground structures have been dismantled and the spent fuel is stored in either the SONGS ISFSI or in the GE-Hitachi Morris facility.

The existing SONGS TSs contain limiting conditions for operation (LCOs) that provide for appropriate functional capability of equipment required for safe operation of the facility, including the plant being in a defueled condition. Since the safety function related to safe storage and management of irradiated fuel at an operating plant is similar to the corresponding function at a permanently defueled facility, the existing TSs provide an appropriate level of control. However, the majority of the existing TSs are only applicable when the reactor is in an operational MODE. Since the SONGS Units 2 and 3, Part 50 licenses no longer authorize emplacement or retention of fuel in the reactor vessels, the LCOs (and associated surveillance requirements (SRs)) that do not apply in a defueled condition are being proposed for deletion. The proposed amendments revise the operating licenses and associated TSs to reflect the permanent cessation of reactor operations and the permanently defueled condition of the reactor vessels at SONGS Units 2 and 3. In general, the changes eliminate those TSs applicable in operating MODES; MODES where fuel is emplaced in the reactor vessel, and certain TSs required for movement of irradiated fuel assemblies. Changes were also proposed to TS definitions, administrative controls, and related to programs and procedures. The proposed amendments also revise the facility operating licenses to clarify or remove certain conditions no longer relevant and add conditions consistent with other permanently shutdown and defueled reactors.

Amendment Nos. 227 and 220 for SONGS Units 2 and 3, respectively, were issued by the NRC on September 30, 2014 (ADAMS Accession No. ML14183B240), to revise certain requirements in the permanently shutdown and defueled facility's TSs, Section 5.0, "Administrative Controls," related to responsibilities, organization, and facility staff qualifications that reflect new staffing and training requirements for operating staff. Issuance of the enclosed amendments, in conjunction with the previously issued TS administrative control amendments, completes the revision to the SONGS permanently shutdown and defueled technical specifications. The TSs being proposed for revision by SCE for incorporation into the SONGS Units 2 and 3 facility operating licenses, referred to by SCE as the Permanently Defueled Technical Specifications (PDTS), have been combined into a single TS that applies to both units. The licensee states that the changes to the facility operating licenses and TSs provide an appropriate level of safety, considering the reduced risk of an offsite radiological release from the remaining postulated design-basis accidents (DBAs) associated with a defueled plant, as described in this safety analysis.

2.0 REGULATORY EVALUATION

2.1 Technical Specifications

Section 182a of the Atomic Energy Act requires applicants for nuclear power plant operating licenses to include TSs as part of the application. The NRC's regulatory requirements related to the content of the TSs are contained in 10 CFR 50.36, "Technical specifications." Pursuant to 10 CFR 50.36, each operating license issued by the Commission includes TSs and includes items in the following categories: (1) safety limits, limiting safety systems settings and control settings, (2) LCOs, (3) SRs, (4) design features, (5) administrative controls, (6) decommissioning, (7) initial notification, and (8) written reports.

Section 50.36 of 10 CFR states, in part, that "safety limits for nuclear reactors are limits upon important process variables that are found to be necessary to reasonably protect the integrity of certain of the physical barriers that guard against the uncontrolled release of radioactivity... Limiting safety system settings for nuclear reactors are settings for automatic protective devices related to those variables having significant safety functions."

Section 50.36 of 10 CFR provides four criteria to define the scope of equipment and parameters to be included in the TS LCOs. These criteria were developed for licenses authorizing operation (i.e., operating reactors) and focused on instrumentation to detect degradation of the reactor coolant system (RCS) pressure boundary, process variables and equipment, design features, or operating restrictions that affect the integrity of fission product barriers during DBAs or transients. A fourth criterion refers to the use of operating experience and probabilistic risk assessment to identify and include in the TSs those structures, systems, and components (SSCs) shown to be significant to public health and safety.

SRs are requirements relating to test, calibration, or inspection to assure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the LCO will be met.

Design features to be included are those features of the facility such as materials of construction and geometric arrangements, which, if altered or modified, would have a significant effect on safety and are not covered in categories described in paragraphs 50.36(c)(1), (2), and (3).

Administrative controls are the provisions relating to organization and management, procedures, recordkeeping, review and audit, and reporting necessary to assure operation of the facility is in a safe manner.

The regulations in 10 CFR 50.36 further state that TSs involving safety limits, limiting safety system settings, and limiting control system settings; LCOs; SRs; design features; and administrative controls for decommissioning facilities will be developed on a case-by-case basis.

A general discussion of the criteria that were used by the NRC staff in its evaluation to ensure that the TS LCOs proposed for deletion are no longer required to be included in the TSs is

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provided below. These criteria were also used in the evaluation of the proposed changes to the existing TSs and the proposed new TSs.

Criterion 1 of 10 CFR 50.36(c)(2)(ii)(A) states that TS LCOs must be established for "installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary [RCPB]." Since no fuel is present in the reactor, maintenance of the RCS pressure boundary as a fission product barrier is no longer relevant at the SONGS Units 2 and 3 facility, and therefore, this criterion is not applicable.

Criterion 2 of 10 CFR 50.36(c)(2)(ii)(B) states that TS LCOs must be established for a "process variable, design feature, or operating restriction that is an initial condition of a design basis accident or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier." The purpose of this criterion is to capture those process variables that have initial values assumed in the DBA and transient analyses, and which are monitored and controlled during power operation. The scope of DBAs applicable to a reactor permanently shutdown and defueled is reduced from those postulated for an operating reactor, and most TSs satisfying Criterion 2 are no longer applicable. The scope of applicable DBAs that apply to SONGS Units 2 and 3 are discussed in more detail in Sections 3.1 through 3.6 of this safety evaluation (SE). There are no transients that continue to apply to the permanently shutdown and defueled reactors.

Criterion 3 of 10 CFR 50.36(c)(2)(ii)(C) states that TS LCOs must be established for a SSC "that is part of the primary success path and which functions or actuates to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier." The intent of this criterion is to capture into TSs those SSCs that are part of the primary success path of a safety sequence analysis. Also captured by this criterion are those support and actuation systems that are necessary for items in the primary success path to successfully function. The primary success path of a safety sequence analysis consists of the combination and sequences of equipment needed to operate (including consideration of the single failure criterion), so that the plant response to DBAs and transients limits the consequences of these events to within the appropriate acceptance criteria. The scope of applicable DBAs that apply to SONGS Units 2 and 3 are discussed in more detail in Sections 3.1 through 3.6 of this SE. There are no transients that continue to apply to the permanently shutdown and defueled reactors.

Criterion 4 of 10 CFR 50.36(c)(2)(ii)(D) states that TS LCOs must be established for SSCs "which operating experience or probabilistic risk assessment has shown to be significant to public health and safety." The intent of this criterion is that risk insights and operating experience be factored into the establishment of TS LCOs. There are no longer any DBAs at SONGS Units 2 and 3 that can result in a significant offsite radiological risk to public health and safety.

The NRC staff notes that in the course of this evaluation, information contained in DRAFT NUREG-1625, "Proposed Standard Technical Specifications for Permanently Defueled Westinghouse Plants," March 1998 (ADAMS Accession No. ML082330233), was also considered. This draft NUREG provides examples of decommissioning TSs for Westinghouse

pressurized water reactors that the staff has previously found acceptable during TS reviews for permanently shutdown and defueled reactors.

2.2 Radiological Consequences from Design-Basis Accidents

During normal power reactor operations, the forced flow of water through the RCS removes the heat generated by the reactor. The RCS, operating at high temperatures and pressures, transfers this heat through the steam generator (SG) tubes to the secondary system. The most severe postulated accidents for nuclear power plants involve damage to the nuclear reactor core and the release of large quantities of fission products to the RCS and subsequent release of some fission products to the environment. Many of the accident scenarios postulated in the facility safety analysis report involve failures or malfunctions of systems that could affect the reactor core. With the termination of reactor operations and the permanent removal of the fuel from the reactor core, such accidents are no longer possible. Therefore, the postulated accidents involving failure or malfunction of the reactor, RCS, or secondary system are no longer applicable. Postulated accidents that could potentially apply to a permanently shutdown and defueled facility include a fuel handling accident (FHA), an accidental release of waste liquid, an accidental release of waste gas, a spent fuel cask drop accident, and a spent fuel pool boiling event. The potential offsite consequences of these events are affected by the time available for decay of fission products in the fuel and, possibly, the availability of engineered safety features, such as ventilation systems to filter fission products from the accident area atmosphere before they are released outside the facility.

The regulations in 10 CFR 50.67, "Accident source term" state, in part, that:

> (i) An individual located at any point on the boundary of the exclusion area for any 2-hour period following the onset of the postulated fission product release, would not receive a radiation dose in excess of 0.25 Sv [Sievert] (25 rem) total effective dose equivalent (TEDE), (ii) An individual located at any point on the outer boundary of the low population zone, who is exposed to the radioactive cloud resulting from the postulated fission product release (during the entire period of its passage), would not receive a radiation dose in excess of 0.25 Sv (25 rem) total effective dose equivalent (TEDE), and (iii) Adequate radiation protection is provided to permit access to and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 0.05 Sv (5 rem) total effective dose equivalent (TEDE) for the duration of the accident.

Appendix A to 10 CFR Part 50, "General Design Criteria (GDC)," Criterion 19--Control room, states, in part:

> A control room shall be provided from which actions can be taken to operate the nuclear power unit safely under normal conditions and to maintain it in a safe condition under accident conditions, including loss-of-coolant accidents. Adequate radiation protection shall be provided to permit access and occupancy of the control room under accident conditions without personnel receiving

> radiation exposures in excess of 5 rem whole body, or its equivalent to any part of the body, for the duration of the accident. Equipment at appropriate locations outside the control room shall be provided (1) with a design capability for prompt hot shutdown of the reactor, including necessary instrumentation and controls to maintain the unit in a safe condition during hot shutdown, and (2) with a potential capability for subsequent cold shutdown of the reactor through the use of suitable procedures.

Regulatory Guide (RG) 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," Revision 0, July 2000 (ADAMS Accession No. ML003716792), provides the methodology for analyzing the radiological consequences of several DBAs to show compliance with 10 CFR 50.67 - Accident source term. Regulatory Guide 1.183 provides guidance to licensees on acceptable application of alternate source term (AST) submittals, including acceptable radiological analysis assumptions for use in conjunction with the AST.

NUREG-0800, "Standard Review Plan [SRP] for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR [Light-Water Reactor] Edition," SRP, Section 15.0.1, "Radiological Consequence Analyses Using Alternative Source Terms," Revision 0, July 2000 (ADAMS Accession No. ML003734190), provides review guidance to the NRC staff for the review of alternative source term amendment requests. SRP Section 15.0.1 states that the NRC reviewer should evaluate the proposed change against the guidance in RG 1.183. As provided in RG 1.183, the dose acceptance criteria for an FHA are a TEDE of 6.3 rem at the exclusion area boundary (EAB) for the worst 2 hours, 6.3 rem at the outer boundary of the low population zone (LPZ), and 5 rem in the control room (CR) for the duration of the accident.

SRP 11.0, Branch Technical Position 11-5, "Postulated Radioactive Release Due to a Waste Gas System Leak or Failure," provides guidance to the NRC staff in assessing the analysis of an accidental release from the waste gas system.

The NRC approved implementation of the AST methodology at SONGS Units 2 and 3, by Amendment Nos. 210 and 202, "San Onofre Nuclear Generating Station, Units 2 and 3 – Issuances of Amendments Re: Full-Scope Implementation of an Alternative Source Term (TAC Nos. MC5495 and MC5496)," dated December 29, 2006 (ADAMS Accession No. ML063400359). These license amendments represent full scope implementation of the AST, as described in RG 1.183.

NRC Regulatory Issue Summary (RIS) 2006-04, "Experience with Implementation of Alternative Source Terms," dated March 7, 2006 (ADAMS Accession No. ML053460347), discusses experiences with analyzing an accident involving a release from off-gas or waste systems. As part of full AST implementation, some licensees have included an accident involving a release from their off-gas or waste system. For this type of accident, licensees have proposed acceptance criteria of 500 millirem (mrem) TEDE. The acceptance criterion for this event is that associated with the dose to an individual member of the public, as described in 10 CFR Part 20, "Standards for Protection Against Radiation." When the NRC revised 10 CFR Part 20 to incorporate a TEDE dose, the offsite dose to an individual member of the public was changed

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from 500 mrem whole body to 100 mrem TEDE. Therefore, a licensee who chooses to implement AST for an off-gas or waste gas system release, as did SCE, should base its acceptance criteria on 100 mrem TEDE. Licensees may also choose not to implement AST for this accident and continue with their existing analysis and acceptance criteria of 500 mrem whole body.

The U.S. Environmental Protection Agency's (EPA's) "Protective Action Guide (PAG) and Planning Guidance for Radiological Incidents," Draft for Interim Use and Public Comment, issued March 2013 (PAG Manual), provides radiological protection criteria for application to all incidents that would require consideration of protective actions, with the exception of nuclear war. This manual provides recommended numerical PAGs for the principal protective actions available to public officials during a radiological incident.

The Nuclear Energy Institute (NEI) document NEI 99-01, "Development of Emergency Action Levels for Non-Passive Reactors," Revision 6, dated November 2012 (ADAMS Accession No. ML12326A805), provides guidance for the development of emergency action levels (EALs) for reactors in a permanently defueled condition. NEI 99-01, Revision 6, was endorsed by the NRC in a letter dated March 28, 2013 (ADAMS Accession No. ML12346A463). NEI 99-01 states that the accident analysis necessary to adopt the permanently defueled EAL scheme must confirm that the source terms and release motive forces are not sufficient to warrant classification of a site area emergency (SAE) or General Emergency, resulting in the maximum classification level of an Alert during an accident. An SAE would be declared for any event where exposure levels beyond the EAB are expected to exceed 10 percent of the EPA PAGs, which are a projected dose of 1 to 5 rem TEDE in four days for sheltering or evacuation of the public, and a projected dose of 5 rem child thyroid dose from radioactive iodine for administration of prophylactic drugs (potassium iodide). Correspondingly, NEI 99-01 established the SAE classification threshold as 100 mrem TEDE or 500 mrem thyroid committed dose equivalent.

3.0 TECHNICAL EVALUATION

3.1 Design-Basis Accident Analysis

SCE described that, with the permanent cessation of reactor operations and permanent removal of fuel from the reactor vessels for SONGS Units 2 and 3, most of the initial conditions, and most of the accident and transient analyses, that were included in Chapter 15 of the SONGS Updated Final Safety Analysis Report (UFSAR) when Units 2 and 3 were authorized to operate, are no longer possible. Therefore, SCE has updated the SONGS UFSAR to reflect that accidents and transients involving the failure or malfunction of fuel within primary containment, the RCS, or the secondary system are no longer applicable. The only DBA scenarios with the potential to result in a radiological release, as described in the UFSAR that are applicable to the permanently shutdown and defueled SONGS Units 2 and 3, are an FHA in the fuel handling building (FHB), a spent fuel cask drop accident, a SFP boiling accident, a liquid radioactive waste system leak or failure, a radioactive release due to liquid tank failures, and an accidental release of waste gas. Because the waste gas decay tanks have been purged of their contents and analyses of liquid tank failures in SONGS UFSAR Section 15.7.3.3.5 describe that no credible liquid release would exceed 10 CFR Part 20 limits, an accidental release of waste gas

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and a liquid tank failure are not relevant at SONGS. The licensee determined that the remaining DBAs would be within relevant regulatory limits, assuming the fuel activity calculated as of August 2013 and without credit for dose consequence mitigation by engineered safety feature (ESF) systems. The NRC staff's technical evaluation of the licensee's analysis of the remaining DBAs at SONGS is provided in Sections 3.2 through 3.6, below.

3.2 Fuel Handling Accident Inside Fuel Building

A revision to the FHA accident analysis was incorporated into the SONGS UFSAR, Section 15.7.3.4, under the provisions of 10 CFR 50.59, "Changes, tests, and experiments," to address the permanently defueled condition. The analysis determined a reasonable time, post-cessation of operations, for movement of fuel from the SFP during which, if an FHA occurs, dose consequences would be within 10 CFR 50.67 and RG 1.183 dose limits. The licensee evaluated the maximum 2-hour TEDE to an individual located at the EAB, and the 30-day TEDE to an individual at the outer boundary of the LPZ and in the CR. The resulting doses in SCE's analyses are less than the RG 1.183 and SRP 15.0.1 dose acceptance criteria, the 10 CFR 50.67 limits, and the EPA PAG levels recommended for protection of the public.

The FHA inside the FHB (FHA-FHB) involves the inadvertent dropping of a fuel assembly during fuel handling operations, and the subsequent rupture of fuel pins in the dropped assembly and any stationary assembly impacted by the dropped assembly. A maximum of 472 fuel rods are assumed to fail, as a result of the drop of a fuel assembly onto the fuel assemblies stored in SFP fuel racks. The fission product inventory in the fuel rod gap of the damaged rods is assumed to be released instantaneously into the SFP. The FHA-FHB dose analysis models 17 months (12,240 hours) of radioactive decay prior to the event. The NRC staff finds that the decay time assumed by the licensee is consistent with RG 1.183, Regulatory Position 3.1, "Fission Product Inventory," which provides that, "For events postulated to occur while the facility is shutdown, e.g., a fuel handling accident, radioactive decay from the time of shutdown may be modeled."

The SFP water level is controlled by current SONGS Units 2 and 3 TS LCO 3.7.16 (renumbered to TS 3.1.1 in the proposed permanently defueled TSs), which limits the movement of irradiated fuel assemblies in the SFP, unless the water level is at least 23 feet over the top of the irradiated fuel assemblies, seated in the storage racks. As such, the licensee assumes that the SFP water level is at least 23 feet over the top of the irradiated fuel assemblies, seated in the storage racks, at the commencement of an FHA-FHB.

Should an FHA occur, fission products released from the damaged fuel are decontaminated by passage through the pool water, with the degree of decontamination dependent upon their physical and chemical forms. The licensee assumed no decontamination for noble gases, a decontamination factor of 200 for radioiodine, and retention of all aerosol and particulate fission products. This is consistent with RG 1.183, Appendix B, Section 2, "Water Depth," which provides that, "If the depth of water above the damaged fuel is 23 feet or greater, the decontamination factors for the elemental and organic species are 500 and 1, respectively, giving an overall effective decontamination factor of 200..."

The radioactive material that escapes from the SFP to the FHB is assumed to be released to the environment over a 2-hour time period. The FHA-FHB dose analysis does not credit the generation of an engineered safety feature actuation system (ESFAS) fuel handling building isolation signal (FHIS). The FHB normal ventilation exhaust is assumed to remain operational throughout the FHA-FHB event. The FHA-FHB AST dose analysis does not model a reduction in the amount of radioactive material available for release from the FHB by the fuel handling building Post-Accident Cleanup Unit (PACU) filter system. Therefore, the licensee assumes the release to the environment is an unfiltered release via the FHB normal ventilation exhaust system through the main plant vent, or as leakage through FHB penetrations. This is consistent with RG 1.183, Appendix B, Section 4.1, which states, "The radioactive material that escapes from the fuel pool to the fuel building is assumed to be released to the environment over a 2-hour time period."

Activity released during the FHA-FHB event is transported by atmospheric dispersion to the CR heating, ventilation and air conditioning (HVAC) intake and to the offsite EAB and LPZ dose receptors. Consistent with RG 1.183, Regulatory Position 5.3, "Meteorology Assumptions," the atmospheric dispersion factor values for the EAB and the LPZ, which were approved by the NRC during initial facility licensing were used by the licensee in performing the AST radiological analyses. The NRC had also approved the use of these meteorology atmospheric dispersion values by Amendment Nos. 210 and 202, dated December 29, 2006 (for SONGS Units 2 and 3, respectively). Consistent with RG 1.183, Regulatory Position 4.1.7, no correction is made for depletion of the effluent plume by deposition on the ground.

The CR dose during a design-basis FHA-FHB, following permanent shutdown of SONGS Units 2 and 3, was based on no credit for the Control Room Emergency Air Cleanup System (CREACUS) and Control Room Isolation Signal (CRIS) and no gamma radiation shine from CREACUS charcoal and high-efficiency particulate air (HEPA) filters. Control room doses are evaluated at various CR unfiltered inflow (including in leakage) flow rates. The flow rates were varied from 500 cubic feet per minute (cfm) to 15,000 cfm, but only the bounding CR dose is reported. The SONGS site-specific 95th percentile meteorology atmospheric dispersion factors for the CR were used.

The licensee concluded that the radiological consequences at the EAB and LPZ and in the CR are within the dose criteria for DBAs specified in 10 CFR 50.67 and SRP Section 15.0.1. The licensee also concluded that the radiological consequences are less than the dose criteria specified in the EPA PAG Manual. The NRC staff reviewed the licensee's evaluation and performed confirmatory calculations. In performing this review, the NRC staff relied upon information provided by the licensee, as well as, NRC staff experience in performing similar reviews. The NRC staff reviewed the methods, parameters, and assumptions that the licensee used in its radiological dose consequence analyses and concludes that they are acceptable because they are consistent with the guidance provided in RG 1.183. Using the FHA-FHB analyses assumptions described above, the NRC staff's confirmatory analyses of the licensee's FHA-FHB yield results for the CR, EAB, and LPZ that are less than the RG 1.183 and SRP 15.0.1 dose acceptance criteria and would not exceed the EPA PAG recommendations at the EAB.

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3.3 Spent Fuel Cask Drop Accident

A re-analysis of the spent fuel cask drop accidents specified in the UFSAR, Section 15.7.3.5, was performed with a cask load of up to 32 fuel assemblies and a minimum of 17 months of decay. The spent fuel cask drop event is evaluated based on the potential of the cask drop to cause a release of radioactive material. This includes consideration of the allowed travel paths of the casks, their lift heights, and the items onto which they can be dropped. Even though single-failure-proof cranes are used at SONGS Units 2 and 3 to lift a spent fuel transfer cask out of a cask pool, a drop can be postulated when the cask is placed on the upper shelf (i.e., step) of a cask pool when performing a yoke-lift change-out, prior to the transfer cask being welded closed. The spent fuel cask drop accident considered to bound the radiological consequences of a spent fuel transfer cask drop (due to a seismic event) is from the upper shelf in the cask pool back into the lower portion of the cask pool. During this postulated accident, the transfer cask is not restrained and could fall back into the lower portion of the cask pool if an earthquake occurs. The fuel rods from all 32 fuel assemblies present in a transfer cask are conservatively assumed to rupture on impact with the bottom of the cask pool. All of the radioactive iodine and noble gases present in the gap volumes of the spent fuel rods are assumed to be released from the unwelded transfer cask. As required by the AST Amendment Nos. 210 and 202, dated December 29, 2006 (for SONGS Units 2 and 3, respectively), the new analysis was performed by the licensee using the AST methodology, including TEDE criteria. The NRC staff concludes that the licensee's modelling of decay time is consistent with RG 1.183, Regulatory Position 3.1.

Other than the number of fuel assemblies considered to fail, the radiological consequence analysis model is identical to that of the FHA in the FHA-FHB (see Section 3.2 of this SE). The fission product inventory in the fuel rod gap of the damaged rods is assumed to be released instantaneously into the SFP. The SFP water level is required to be at least 23 feet over the top of the irradiated fuel assemblies seated in the storage racks, as controlled by TSs. Consistent with RG 1.183, Appendix B, Regulatory Position 4.1, the radioactive material that escapes from the SFP to the FHB is released to the environment over a 2-hour time period, ensuring that at least 99.9 percent of the gaseous activity will be released to the environment. Consistent with RG 1.183, Regulatory Position 5.3, the atmospheric dispersion factor values for the EAB and the LPZ that were approved by the NRC during initial facility licensing are used in performing the AST radiological analyses. The NRC had also approved use of these meteorology atmospheric dispersion values by Amendment Nos. 210 and 202 for SONGS Units 2 and 3, respectively. Consistent with RG 1.183, Regulatory Position 4.1.7, no correction is made for depletion of the effluent plume by deposition on the ground.

The licensee concluded that the radiological consequences at the EAB and LPZ and in the CR are within the dose criteria for the DBAs, as specified in 10 CFR 50.67. The licensee also concluded that the radiological consequences are less than the dose criteria specified in the EPA PAG Manual. The NRC staff reviewed the licensee's evaluation and performed confirmatory calculations. In performing this review, the NRC staff relied upon information provided by the licensee and NRC staff experience in performing similar reviews. The NRC staff reviewed the methods, parameters, and assumptions that the licensee used in its radiological dose consequence analyses and concludes that they are acceptable because they are consistent with the guidance provided in RG 1.183. Using the analyses assumptions

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described above, the NRC staff's confirmatory analyses yielded results for the CR, EAB, and LPZ that were less than the RG 1.183 dose acceptance criteria and would not have exceeded the EPA PAG recommendations at the EAB.

3.4 Spent Fuel Pool Boiling Accident

The postulated loss of all SFP cooling is assumed to result in SFP boiling and release of a portion of the radionuclide inventory contained in the stored spent fuel assemblies and the SFP water. The re-evaluation of the radiological consequences for the SFP boiling event assumes a minimum of 17 months since the shutdown of SONGS Units 2 and 3. The licensee used the AST methodology in performing this evaluation. The NRC staff concludes that the licensee's modelling of decay time is consistent with RG 1.183, Regulatory Position 3.1.

The radiological consequence analysis does not differentiate between the activity release rates before and after the onset of SFP boiling. Noble gas, iodine and tritium activity present in the failed fuel rod gap spaces of fuel rods, stored within the SFP, is released to the SFP water at the noble gas, iodine, and tritium escape rate coefficients, with the added conservatism of an assumed spiking factor of 100. The noble gas and iodine fuel rod gap fractions are consistent with the AST methodology. The tritium fuel rod gap fraction is assumed to be the same as that for the majority of noble gas and iodine isotopes. Tritium activity present in the SFP water prior to the loss of SFP cooling, is assumed to be released at the SFP boiling rate for the duration of the event. Both before and after the onset of SFP boiling spent fuel noble gases, iodine and tritium gas escaping from the failed fuel rod gap spaces are assumed to be instantaneously released with no hold up or iodine partitioning in the SFP water. The SFP boiling rate is a function of the decay heat load and the heat of vaporization of water.

Following a loss of SFP cooling, activity releases from the spent fuel due to evaporation and boiling disperse to the CR, EAB, and LPZ locations. No credit is taken for activity retention within the FHB. No credit is taken for FHIS or filtration by the FHB PACUs. All activity escaping from the SFP is assumed to be instantaneously released to the environment and atmospherically dispersed to the CR and offsite dose receptors. No credit is taken for CRIS or CREACUS.

The SFP boiling accident consequence analysis uses the identical model used for the FHA-FHB (see Section 3.2 of this SE). Consistent with RG 1.183, Appendix B, Section 4.1, the radioactive material that escapes from the SFP to the FHB is released to the environment over a 2-hour time period, ensuring that at least 99.9 percent of the gaseous activity will be released to the environment. For conservatism, the CR dose is calculated for an individual at the CR outside air intake location.

Consistent with RG 1.183, Regulatory Position 5.3, the atmospheric dispersion factor values for the EAB and the LPZ, which were approved by the NRC during initial facility licensing, are used in performing the AST radiological analyses. The NRC staff had also approved the use of these meteorology atmospheric dispersion values by Amendment Nos. 210 and 202, dated December 29, 2006 (for SONGS Units 2 and 3, respectively). Consistent with RG 1.183,

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Regulatory Position 4.1.7, no correction is made for depletion of the effluent plume by deposition on the ground.

The licensee concluded that the radiological consequences at the EAB, LPZ, and CR are within the dose criteria for DBAs, as specified in 10 CFR 50.67. The licensee also concluded that the radiological consequences are less than the dose criteria specified in the EPA PAG Manual. The NRC staff reviewed the licensee's evaluation and performed confirmatory calculations. In performing this review, the NRC staff relied upon information provided by the licensee, as well as NRC staff experience in performing similar reviews. The NRC staff reviewed the methods, parameters, and assumptions that the licensee used in its radiological dose consequence analyses and finds that they are acceptable because they are consistent with the guidance provided in RG 1.183. Using the analyses assumptions described above, the NRC staff's confirmatory analyses yielded results for the CR, EAB, and LPZ that are less than the RG 1.183 dose acceptance criteria and would not exceed the EPA PAG recommendations at the EAB.

3.5 Radioactive Waste System Leak or Failure (Release to Atmosphere) Accident

The radioactive waste system leak or failure (with release to atmosphere) accident analysis (UFSAR Section 15.7.3.2) was revised to calculate the EAB and LPZ doses using the AST methodology. As required by the AST Amendment Nos. 210 and 202 for SONGS Units 2 and 3, respectively, the evaluation includes TEDE dose criteria, and a revised offsite dose acceptance criterion of 100 mrem TEDE, as addressed in NRC RIS 2006-04. The evaluation does not assume any post-shutdown decay time.

Releases from the Liquid Radioactive Waste System considered rupture of: radwaste tanks, refueling water storage tanks, primary ion-exchangers, and the blowdown demineralizer neutralization sump line. The most limiting of these is defined as an unexpected and uncontrolled release of the radioactive liquid stored in a radwaste secondary tank. The radwaste secondary tanks are Seismic Category II, Quality Class III tanks at atmospheric pressure. Rupture of these tanks is considered a limiting fault. A radwaste secondary tank rupture would release the liquid contents into the auxiliary building (radwaste area). It is assumed that all of the radioactive fission gases and iodines are released to the outside atmosphere within 2 hours.

The dose analysis for persons located at the EAB and the LPZ considers the dose consequences of inhalation and submersion in a radioactive cloud, as described in RG 1.183. Activity released during the event is transported by atmospheric dispersion to the offsite EAB and LPZ dose receptors. Consistent with RG 1.183, Regulatory Position 5.3, the atmospheric dispersion factor values for the EAB and the LPZ, which were approved by the NRC during initial facility licensing, are used in performing the AST radiological analyses. The NRC staff had also approved use of these meteorology atmospheric dispersion values by Amendment Nos. 210 and 202 for SONGS Units 2 and 3, respectively. Consistent with RG 1.183, Regulatory Position 4.1.7, no correction is made for depletion of the effluent plume by deposition on the ground.

The licensee concluded that the radiological consequences are less than 100 mrem TEDE offsite dose criterion per RIS 2006-04. The licensee also concluded that the radiological consequences are less than the dose criteria specified in the EPA PAG Manual. The NRC staff reviewed the licensee's evaluation and performed confirmatory calculations. In performing this review, the NRC staff relied upon information provided by the licensee, as well as NRC staff experience in performing similar reviews. The NRC staff reviewed the methods, parameters, and assumptions that the licensee used in its radiological dose consequence analyses and concludes that they are acceptable because they are consistent with the guidance provided in RG 1.183. Using the analyses assumptions described above, the NRC staff's confirmatory analyses yielded results for the EAB, LPZ, and CR that are less than RG 1.183 dose acceptance criteria and are also less than the offsite dose criteria per RIS 2006-04 and would not exceed the EPA PAG recommendations at the EAB.

3.6 Accident Analysis Conclusions

The NRC staff reviewed the assumptions, inputs, and methods used by the licensee to assess the radiological impacts of the proposed changes. The NRC staff finds that the licensee's analyses are acceptable because their analysis methods and assumptions are consistent with the guidance of RG 1.183. The NRC staff compared the doses estimated by the licensee to the applicable criteria and to the results of confirmatory analyses by the staff. The NRC staff finds, with reasonable assurance, that the licensee's estimates of the EAB, LPZ, and CR doses due to postulated DBAs at SONGS will comply with the requirements of 10 CFR 50.67 and the guidance of RG 1.183, and 10 CFR Part 20, as addressed in NRC RIS 2006-04. The NRC staff finds, with respect to the consequences of the remaining DBAs at SONGS, that no CR dose limits will be exceeded and that any offsite radiological release will not exceed the EPA PAGs at the EAB.

3.7 Proposed TS Changes

3.7.1 Section 1.1, Definitions

The licensee proposed deleting the following definitions because they pertain to an operating reactor. Since SONGS Units 2 and 3 are permanently shut down and defueled, the definitions have no relevance and no longer apply:

> AXIAL SHAPE INDEX (ASI) – ASI shall be the power generated in the lower half of the core less the power generated in the upper half of the core, divided by the sum of the power generated in the lower and upper halves of the core.
>
> $$\mathrm{ASI} = \frac{\text{lower}-\text{upper}}{\text{lower}+\text{upper}}$$
>
> AZIMUTHAL POWER TILT (T_q) - AZIMUTHAL POWER TILT shall be the power asymmetry between azimuthally symmetric fuel assemblies.
>
> CHANNEL CALIBRATION – A CHANNEL CALIBRATION shall be the adjustment, as necessary, of the channel output such that it responds within the

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necessary range and accuracy to known values of the parameter that the channel monitors. The CHANNEL CALIBRATION shall encompass the entire channel, including the required sensor, alarm, display, and trip functions, and shall include the CHANNEL FUNCTIONAL TEST. Calibration of instrument channels with resistance temperature detector (RTD) or thermocouple sensors may consist of an inplace cross calibration of the sensing elements and normal calibration of the remaining adjustable devices in the channel. Whenever a sensing element is replaced, the next required inplace cross calibration consists of comparing the other sensing elements with the recently installed sensing element.

The CHANNEL CALIBRATION may be performed by means of any series of sequential, overlapping, or total channel steps so that the entire channel is calibrated.

CHANNEL CHECK – A CHANNEL CHECK shall be the qualitative assessment, by observation, of channel behavior during operation. This determination shall include, where possible, comparison of the channel indication and status to other indications or status derived from independent instrument channels measuring the same parameter.

CHANNEL FUNCTIONAL TEST - A CHANNEL FUNCTIONAL TEST shall be:

a. Analog channels - the injection of a simulated or actual signal into the channel as close to the sensor as practicable to verify OPERABILITY, including required alarms, interlocks, display and trip functions;

b. Bistable channels (e.g., pressure switches and switch contacts) - the injection of a simulated or actual signal into the channel as close to the sensor as practicable to verify OPERABILITY, including required alarm and trip functions; or

c. Digital computer channels - the use of diagnostic programs to test digital computer hardware and the injection of simulated process data into the channel to verify OPERABILITY, including alarm and trip functions.

The CHANNEL FUNCTIONAL TEST may be performed by means of any series of sequential, overlapping, or total channel steps so that the entire channel is tested.

CORE ALTERATION - CORE ALTERATION shall be the movement or manipulation of any fuel, sources, reactivity control components, or other components, excluding control element assemblies (CEAs) withdrawn into the upper guide structure, affecting reactivity, within the reactor vessel with the vessel head removed and fuel in the vessel. Suspension of CORE

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ALTERATIONS shall not preclude completion of movement of a component to a safe position.

CORE OPERATING LIMITS REPORT (COLR) - The COLR is the unit specific document that provides cycle specific parameter limits for the current reload cycle. These cycle specific parameter limits shall be determined for each reload cycle in accordance with Specification 5.7.1.5. Plant operation within these limits is addressed in individual Specifications.

DOSE EQUIVALENT I-131 - DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcuries/gram) that alone would produce the same thyroid dose as the quantity and isotopic mixture of 1-131, 1-132, 1-133, 1-134, and 1-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in ICRP-30, Supplement to Part 1, page 192-212, Table titled, "Committed Dose Equivalent in Target Organs or Tissues per Intake of Unit Activity."

$\bar{E}$- AVERAGE DISINTEGRATION ENERGY - $\bar{E}$ shall be the average (weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling) of the sum of the average beta and gamma energies per disintegration (in MeV) for isotopes, other than iodines, with half lives > 15 minutes, making up at least 95% of the total noniodine activity in the coolant.

ENGINEERED SAFETY FEATURE (ESF) RESPONSE TIME - The ESF RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its ESF actuation setpoint at the channel sensor until the ESF equipment is capable of performing its safety function (i.e., the valves travel to their required positions, pump discharge pressures reach their required values, etc.). Times shall include diesel generator starting and sequence loading delays, where applicable. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured. In lieu of measurement, response time may be verified for selected components provided that the components and methodology for verification have been previously reviewed and approved by the NRC.

LEAKAGE – LEAKAGE shall be:

a. Identified LEAKAGE

   1. LEAKAGE, such as that from pump seals or valve packing (except reactor coolant pump (RCP) leakoff), that is captured and conducted to collection systems or a sump or collecting tank;

   2. LEAKAGE into the containment atmosphere from sources that are both specifically located and known either not to interfere with the

operation of leakage detection systems or not to be pressure boundary LEAKAGE; or

3. Reactor Coolant System (RCS) LEAKAGE through a steam generator to the Secondary System (primary to secondary LEAKAGE).

b. Unidentified LEAKAGE

All LEAKAGE that is not identified LEAKAGE.

c. Pressure Boundary LEAKAGE

LEAKAGE (except primary to secondary LEAKAGE) through a nonisolable fault in an RCS component body, pipe wall, or vessel wall.

MODE – A MODE shall correspond to any one inclusive combination of core reactivity condition, power level, average reactor coolant temperature, and reactor vessel head closure bolt tensioning specified in Table 1.1-1 with fuel in the reactor vessel.

PHYSICS TESTS – PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation. These tests are:

a. Described in Chapter 14, “Initial Test Program of the SONGS Units 2 and 3 UFSAR;

b. Authorized under the provisions of 10 CFR 50.59; or

c. Otherwise approved by the Nuclear Regulatory Commission.

RCS PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR) - The PTLR is the unit specific document that provides the reactor vessel pressure and temperature limits, including heatup and cooldown rates, for the current reactor vessel fluence period. These pressure and temperature limits shall be determined for each fluence period in accordance with Specification 5.7.1.6.

RATED THERMAL POWER (RTP) – RTP shall be a total reactor core heat transfer rate to the reactor coolant of 3438 MWt.

REACTOR PROTECTION SYSTEM (RPS) RESPONSE TIME - The RPS RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its RPS trip setpoint at the channel sensor until electrical power to the CEAs drive mechanism is interrupted. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire

response time is measured. In lieu of measurement, response time may be verified for selected components provided that the components and methodology for verification have been previously reviewed and approved by the NRC.

SHUTDOWN MARGIN (SDM) – SDM shall be the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming:

a. All full length CEAs (shutdown and regulating) are fully inserted except for the single CEA of highest reactivity worth, which is assumed to be fully withdrawn. However, with all CEAs verified fully inserted by two independent means, it is not necessary to account for a stuck CEA in the SDM calculation. With any CEAs not capable of being fully inserted, the reactivity worth of these CEAs must be accounted for in the determination of SDM, and

b. There is no change in part length CEA position.

STAGGERED TEST BASIS - A STAGGERED TEST BASIS shall consist of the testing of one of the systems, subsystems, channels, or other designated components during the interval specified by the Surveillance Frequency, so that all systems, subsystems, channels, or other designated components are tested during *n* Surveillance Frequency intervals, where *n* is the total number of systems, subsystems, channels, or other designated components in the associated function.

THERMAL POWER – THERMAL POWER shall be the total reactor core heat transfer rate to the reactor coolant.

In conjunction with deletion of the term "MODE," TS Table 1.1-1, "MODES," is also being deleted.

The NRC staff examined the TS definitions proposed for deletion and concluded that all the terms listed above are only meaningful to a reactor authorized to operate. Since SONGS Units 2 and 3 are permanently shut down and defueled, the NRC staff finds that the licensee's proposed change to delete these definitions from the TSs is acceptable.

In addition, the licensee proposed adding a definition for CERTIFIED FUEL HANDLER. The licensee proposed to define a certified fuel handler as:

CERTIFIED FUEL HANDLER – A CERTIFIED FUEL HANDLER is an individual who complies with provisions of the CERTIFIED FUEL HANDLER training program required by TS 5.3.2.

TS 5.3.2 states, "An NRC approved training and retraining program for the Certified Fuel Handlers shall be maintained." The NRC staff finds the definition of a Certified Fuel Handler

conforms to the usage contained in the Administrative Controls section of the SONGS Units 2 and 3 permanently defueled TSs and is consistent with the definition in 10 CFR Part 50 and is, therefore, acceptable.

3.7.2 Section 1.2, Logical Connectors

Section 1.2, "Logical Connectors," of the SONGS TSs provides an explanation of the use of logical connectors. Logical connectors are used in TSs to discriminate between, and yet connect, discrete Conditions, Required Actions, Completion Times, Surveillances, and Frequencies. The only logical connectors that appear in TSs are AND and OR. The physical arrangement of these connectors constitutes logical conventions with specific meanings.

The licensee proposed deleting Example 1.2-2 in this section of the SONGS Units 2 and 3 defueled TSs. Example 1.2-2 explains the use of multiple logical connectors and nested connectors in the Required Action section. The licensee states that Example 1.2-1 adequately explains the use of the logical connectors that remain in the proposed defueled TS.

The only logical connector retained in the defueled TS is contained in the Required Action section for renumbered LCO 3.1.2. This use of logical connectors is fully illustrated by Example 1.2-1 in Section 1.2. Therefore, the NRC staff finds that the licensee's proposed change to delete Example 1.2-2 from TS Section 1.2 is acceptable.

3.7.3 Section 1.3, Completion Times

Section 1.3, "Completion Times," of the SONGS TSs establishes the completion time convention throughout the TSs and provides guidance for its use. The licensee has proposed to replace each reference to "operation of the unit" and "unit" with the new terminology, "storage of fuel assemblies" and "facility," respectively, since operation of the unit is no longer permitted and safe storage of fuel assemblies is the primary objective of the permanently defueled TSs. In its February 23, 2015, response to the NRC staff's request for additional information (RAI), the licensee stated that the requirements in the defueled TSs are applicable to the storage of any fuel assembly, and are not limited to the safe storage of irradiated fuel assemblies. In addition, the licensee proposed to delete references to "MODE" to be consistent with the removal of these definitions from TSs and because this term is no longer used in the Required Actions of the subsequent remaining LCOs in the proposed SONGS Units 2 and 3 defueled TSs. The licensee also proposed to delete Examples 1.3-2 through 1.3-7 because these examples refer to activities that no longer pertain to a permanently defueled condition.

The proposed change is shown below, with a strikethrough of the current wording and highlighting of the proposed changes:

BACKGROUND

Limiting Condition [sic] for Operation (LCOs) specify minimum requirements for ensuring safe ~~operation of the unit~~ storage of fuel assemblies. The ACTIONS associated with an LCO state Conditions that typically describe the ways in which

the requirements of the LCO can fail to be met. Specified with each stated Condition are Required Action(s) and Completion Time(s).

DESCRIPTION

The Completion Time is the amount of time allowed for completing a Required Action. It is referenced to the time of discovery of a situation (e.g., inoperable equipment or variable not within limits) that requires entering an ACTIONS Condition unless otherwise specified, providing the ~~unit~~ facility is in a ~~MODE or~~ specified condition stated in the Applicability of the LCO. Required Actions must be completed prior to the expiration of the specified Completion Time. An ACTIONS Condition remains in effect and the Required Actions apply until the Condition no longer exists or the ~~unit~~ facility is not within the LCO Applicability. ...

The licensee proposed to delete several paragraphs of the Description Section that discuss entry into multiple Conditions, subsequent discovery of additional inoperable equipment, and the effects on the total Completion Times.

The licensee proposed to modify the Example, as follows:

EXAMPLE~~S~~ The following example~~s~~ illustrates the use of Completion Times ~~with different types of Conditions and changing Conditions~~.

EXAMPLE 1.3-1

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Required Action and associated Completion Time not met.	B.1 ~~Be in MODE 3~~ Verify....	6 hours
	AND	
	B.2 ~~Be in MODE 5~~ Restore....	36 hours

Condition B has two Required Actions. Each Required Action has its own separate Completion Time. Each Completion Time is referenced to the time that Condition B is entered.

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> The Required Actions of Condition B are to ~~be in MODE 3~~ perform the verification within 6 hours AND perform the restoration ~~in MODE 5~~ within 36 hours. A total of 6 hours is allowed for performing the verification ~~reaching MODE 3~~ and a total of 36 hours (not 42 hours) is allowed for performing the restoration ~~reaching MODE 5~~ from the time that Condition B was entered. If verification is performed ~~MODE 3 is reached~~ within 3 hours, the time allowed for ~~reaching MODE 5~~ performing the restoration is the next 33 hours because the total time allowed for performing the restoration ~~reaching MODE 5~~ is 36 hours.

The NRC staff has reviewed the proposed wording changes to the TS 1.3 Completion Times guidance and has determined that they are consistent with the transition from an operating reactor to a permanently shutdown and defueled facility with a primary safety focus of storage of fuel assemblies. The proposed changes also remove references to operating MODES that are no longer permitted following the licensee's submittal of certifications in accordance with 10 CFR 50.82(a)(2).

Examples 1.3-2 through 1.3-7, which are proposed to be deleted, provide an explanation of the time requirements for transitioning into a MODE in which the requirements are not applicable associated with entry in TS 3.0.3, or provide an explanation of more complex arrangements of Required Actions and Completion Times beyond those retained in the defueled TSs. TS 3.0.3 is being deleted as discussed in Section 3.7.6 of this safety evaluation. The NRC staff finds that these examples are no longer necessary to understand and properly implement the remaining Required Actions and Completion Times.

For the reasons discussed above, the NRC staff has determined that the revision to Example 1.3-1 and the deletion of the remaining examples are appropriate. Therefore, the NRC staff finds that the licensee's proposed changes to TS Section 1.3 are acceptable.

3.7.4 Section 1.4, Frequency

Section 1.4, "Frequency," of the SONGS TSs, defines the proper use and application of Frequency requirements throughout the TSs. In this section, the licensee has proposed to delete the final paragraph in the description section. The final paragraph of the TS 1.4 description section discusses notes that modify the frequency of performance of some surveillances and the applicability of operating MODE entry restrictions of SR 3.0.4. None of the surveillances in the proposed TSs contain notes that modify the frequency of performance or the conditions during which the acceptance criteria must be satisfied. Therefore, this paragraph is not applicable to the proposed TS LCOs or SRs and may be deleted. Specifically, the following is being deleted:

> Situations where a Surveillance could be required (i.e., its Frequency could expire), but where it is not possible or not desired that it be performed until sometime after the associated LCO is within its Applicability, represent potential

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SR 3.0.4 conflicts. To avoid these conflicts, the SR (i.e., the Surveillance or the Frequency) is stated such that it is only "required" when it can be and should be performed. With an SR satisfied, SR 3.0.4 imposes no restriction.

The licensee proposed replacing the statement introducing the examples from "In these examples, the Applicability of the LCO ...is MODES 1, 2, and 3" with the statement "In these examples, the Applicability of the LCO...occurs whenever any fuel assembly is stored in the fuel storage pool." The reference to MODEs 1, 2, and 3 is no longer meaningful to the permanently shutdown and defueled condition at SONGS Units 2 and 3.

The licensee proposed to modify Examples 1.4-1 and 1.4-2 to be applicable to a facility that is permanently shutdown and defueled and proposed to delete Example 1.4-3, in its entirety.

A summary of the proposed changes is shown below, with a strikethrough of the current wording and highlighting of the proposed changes:

EXAMPLE 1.4-1

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
~~Perform CHANNEL CHECK~~. Verify. . .	~~12 hours~~ 7days

The licensee proposed to modify the discussion of Example 1.4-1 to reflect the 7 day frequency chosen for the example; to replace the word "unit" with the word "facility"; to delete the reference to MODEs; and to delete the reference to Example 1.4-3, which is also being deleted.

EXAMPLE 1.4-2

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
~~Verify flow is within limits.~~ Verify...	Prior to moving a fuel assembly. . . ~~Once within 12 hours after~~

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SURVEILLANCE	FREQUENCY
	~~≥ 25% RTP~~ ~~AND~~ ~~24 hours thereafter~~

The licensee proposed to revise the discussion of Example 1.4-2, as follows:

> Example 1.4-2 illustrates ~~has two Frequencies. The first is~~ a one time performance Frequency. ~~and the second is of the type shown in Example 1.4-1. The logical connector "AND" indicates that both Frequency requirements must be met. Each time reactor power is increased from a power level < 25% RTP to ≥ 25% RTP, the Surveillance must be performed within 12 hours.~~
>
> ~~The use of "once" indicates a single performance will satisfy the specified Frequency (assuming no other Frequencies are connected by "AND").~~ This type of Frequency does not qualify for the 25% extension allowed by SR 3.0.2. ~~"Thereafter" indicates future performances must be established per SR 3.0.2, but only after a specified condition is first met (i.e., the "once" performance in this example). If reactor power decreases to < 25% RTP, the measurement of both intervals stops. New intervals start upon reactor power reaching 25% RTP~~.

The licensee also proposed to delete Example 1.4-3 because it is not needed in a permanently defueled condition. The example used a reference to operating reactor MODES that are no longer permitted at SONGS Units 2 and 3. Specifically, Example 1.4-3 refers to surveillances to be performed after power is greater than or equal to (≥) 25 percent rated thermal power (% RTP) and discusses MODE entry restrictions.

The licensee stated that the remaining Examples 1.4-1 and 1.4-2 (as revised) are sufficient to explain the application of TS frequency requirements for the permanently defueled SONGS Units 2 and 3, TSs.

The NRC staff has reviewed the proposed changes to TS Section 1.4 and has determined that they are appropriate for a permanently shutdown and defueled reactor. The proposed changes remove references to operating MODES or rated thermal power that are no longer permitted, following certification under the provisions of 10 CFR 50.82(a)(2). The deletion of the surveillance note referring to MODE entry restrictions of SR 3.0.4 is also appropriate since none of the surveillances in the proposed remaining defueled TSs contain notes that modify the

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frequency of performance or the conditions during which the acceptance criteria must be satisfied. Therefore, the NRC staff finds that the licensee's proposed changes to TS Section 1.4 are acceptable.

3.7.5 Section 2.0, Safety Limits

Section 2.0, "Safety Limits," of the SONGS TSs, establishes safety limits (SLs), which are limits upon important process variables that are found to be necessary to reasonably protect the integrity of certain of the physical barriers that guard against the uncontrolled release of radioactivity. SLs ensure that specified acceptable fuel design limits are not exceeded during steady state operation, normal operation transients, and anticipated operational occurrences (AOOs). The SL on RCS pressure protects the integrity of the RCS against overpressurization. In the event of fuel cladding failure, fission products are released into the reactor coolant. The RCS then serves as the primary barrier in preventing the release of fission products into the atmosphere.

TS 2.1, "Safety Limits" (SLs), contains two separate specifications:

- TS 2.1.1, Reactor Core SLs; and
- TS 2.1.2, Reactor Coolant System Pressure SL

TS 2.2, "SL Violations," direct actions to be taken if an SL specified in TS 2.1 is violated.

The restrictions of the SLs defined in TS 2.1.1 prevent overheating of the fuel cladding, and possible cladding perforation, which could result in the release of fission products to the reactor coolant. TS 2.1.1 is applicable in MODES 1 and 2. TS 2.1.2 defines requirements on parameters to protect the integrity of the RCS against overpressure. TS 2.1.2 is applicable in MODES 1, 2, 3, 4, and 5.

The licensee proposed to delete the SLs specified in Section 2.0, because they are not applicable to the permanently shutdown and defueled status of the plant. The licensee stated that the SL TSs limit important process variables that are necessary to reasonably protect the integrity of certain physical barriers required for safe operation of the reactor in MODES 1 through 5. However, 10 CFR 50.82(a)(2) prohibits operation of the reactor or placing fuel in the reactor vessel. Therefore, the SL TSs only address specific process variables that are no longer applicable to SONGS Units 2 and 3.

The NRC staff examined the SLs and their TS Bases. There are three SLs in Section 2.0: a minimum limit on the departure from nucleate boiling ratio (DNBR); a maximum limit on the peak fuel centerline temperature to ensure fuel and cladding integrity; and a maximum RCS pressure to ensure RCS integrity. As stated in the "Bases for DNBR," a limit is placed on the DNBR, such that, no fuel clad damage would occur as a result of normal operation and AOOs. A limit is placed on peak fuel centerline temperature, such that, a hot fuel pellet in the core will not experience centerline fuel melting. The TS Bases for the maximum RCS pressure state that RCS integrity is an important barrier in the prevention of an uncontrolled release of fission products. Because SONGS Units 2 and 3 have permanently shut down and defueled, and the

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licensee has submitted certifications under the provisions of 10 CFR 50.82(a)(2), placing fuel in the reactor vessel and resuming power operations are no longer authorized, therefore, SLs associated with the RCS are no longer applicable. In this condition, there will be no DNBR or peak fuel centerline temperature to be monitored. Based on these findings, the NRC staff concludes the SLs no longer apply. Since the SLs are no longer applicable, TS 2.2, which specifies the actions to be taken if a SL is violated, is no longer necessary. Therefore, the NRC staff finds the licensee's proposed changes to delete TSs 2.1 and 2.2 is acceptable.

3.7.6 Section 3.0, Limiting Condition for Operation and Surveillance Requirement Applicability

Section 3.0, "Limiting Condition for Operation (LCO) Applicability," and "Surveillance Requirement (SR) Applicability," of the SONGS TSs, contains the general requirements applicable to all LCOs and SRs and applies at all times unless otherwise stated in TSs.

LCO 3.0.1, establishes the applicability statement within each individual TS as the requirement for when the LCO shall be met. The licensee proposed to delete the reference to "MODES" and the reference to LCO 3.0.7. The licensee stated that reference to MODES is no longer relevant since SONGS Units 2 and 3 are permanently shut down and defueled. In addition, the deletion of reference to LCO 3.0.7 conforms to the request to delete this LCO from the SONGS Units 2 and 3 TSs, as discussed below.

LCO 3.0.2 establishes that upon discovery of a failure to meet an LCO, the Required Actions of the associated Conditions shall be met. The licensee proposed to delete the references to LCO 3.0.5 and 3.0.6 to conform to the request to delete TS LCO 3.0.5 and 3.0.6 from the SONGS Units 2 and 3 TSs, as discussed below.

LCO 3.0.3 establishes the Actions that must be implemented when an LCO is not met and the associated Actions are not met, or an associated Action is not provided. LCO 3.0.3 requires placing the unit in a MODE or other specified condition in which the LCO does not apply. The licensee proposed to delete LCO 3.0.3, in its entirety, since it no longer applies. The regulations prohibit operation of the plant or placing fuel in the reactor vessel and references to operating MODES is no longer relevant.

LCO 3.0.4 establishes limitations on changing MODES or other specified conditions in the applicability when an LCO is not met. The licensee proposed to delete LCO 3.0.4 since SONGS Units 2 and 3 are permanently shut down and defueled and references to operating MODES are no longer relevant.

LCO 3.0.5 establishes the allowance for restoring equipment to service under administrative controls when it has been removed from service or declared inoperable to comply with Actions. The licensee proposed to delete LCO 3.0.5, in its entirety. The licensee stated that LCO 3.0.5 is no longer necessary because the proposed defueled TSs do not contain requirements for declaring equipment inoperable or removing equipment from service.

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LCO 3.0.6 establishes an exception to LCO 3.0.2 to allow the use of the Safety Function Determination Program. The licensee has proposed to delete that the Safety Function Determination Program, described in TS 5.5.13. The deletion of the Safety Function Determination Program is discussed in Section 3.7.17.4 of this SE. Consequently, the licensee has proposed to delete LCO 3.0.6, in its entirety, to conform the TSs to the proposed deletion of the Safety Function Determination Program.

LCO 3.0.7 pertains to certain reactor physics special tests and operations required to be performed at various times over the life of the unit. The licensee proposed to delete LCO 3.0.7, in its entirety, since reactor physics testing is no longer relevant to a permanently shutdown and defueled facility.

SR 3.0.1 establishes the requirement that SRs must be met during the MODES or other specified conditions in the applicability for which the requirements of the LCO apply, unless otherwise specified in the individual SRs. This specification ensures that surveillances are performed to verify the operability of systems and components, and that variables are within specified limits. The licensee proposed to revise SR 3.0.1 to delete the reference to "MODES" since the reference to "MODES" is no longer relevant for the permanently shutdown and defueled condition at SONGS Units 2 and 3.

SR 3.0.2 permits a 25 percent extension of the interval specified in the Frequency, and establishes the requirements for meeting the specified Frequency for Surveillances and any Required Action with a Completion Time that requires the periodic performance of the Required Action on a "once per…" interval. The licensee proposed to delete the statements that pertain to Frequencies specified as "once" or "once per" and to delete the statement that "[e]xceptions to this Specification are stated in the individual Specifications." The licensee stated that the proposed defueled TSs no longer contain this type of Frequency or Completion Time.

SR 3.0.4 establishes the requirement that all applicable SRs must be met before entry into a MODE or other specified condition in the Applicability. The licensee proposed to delete the reference to "MODE" since SONGS Units 2 and 3 is permanently shut down and defueled and operating MODES are no longer relevant.

The NRC staff has reviewed the proposed changes to TS LCO 3.0.1, SR 3.0.1, and SR 3.0.4 and determined that the changes acceptably removes the reference to "MODE," which is no longer applicable to a permanently shutdown and defueled facility. The staff has also reviewed the change to LCO 3.0.2 and determined that the change acceptably removes the reference to other TS requirements that are being deleted. Therefore, the NRC staff finds that the licensee's proposed changes to these LCOs and SRs are acceptable.

The NRC staff has reviewed the proposed deletion of TS LCOs 3.0.3 and LCO 3.0.4, and has determined that, consistent with the transition to a permanently shutdown and defueled facility, these LCOs are no relevant. Since 10 CFR 50.82(a)(2) prohibits the licensee from operating the plant or placing fuel in the reactor vessel, the references to "MODE," and the discussions about shutting down the unit, are no longer applicable. The NRC staff finds the licensee's

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proposed changes to delete these LCOs, reflect the current SONGS Units 2 and 3 plant status, and, therefore, are acceptable.

The NRC staff has reviewed the proposed deletion of TS LCO 3.0.5. The NRC staff agrees that this specification is no longer necessary because the defueled TSs do not contain requirements to declare equipment inoperable or to remove equipment from service. Since LCO 3.0.5 is being deleted, the deletion of reference to LCO 3.0.5 in LCO 3.0.2 is also appropriate. The NRC staff finds this change appropriately reflects the requirements in the defueled TSs.

The NRC staff has reviewed the proposed deletion of TS LCO 3.0.6. The NRC staff agrees that there will not be any systems in the permanently defueled TSs that are interrelated with other systems that have TS LCOs (support and supported systems). As a result, the conditions of LCO 3.0.6 no longer apply. Therefore, the staff finds that it is appropriate to delete LCO 3.0.6. Since LCO 3.0.6 is being deleted, the deletion of reference to LCO 3.0.6 in LCO 3.0.2 is also acceptable.

The NRC staff has reviewed the proposed deletion of TS LCO 3.0.7. The facility operating licenses no longer permit emplacement of fuel in the reactor vessels or operation of the facility, and therefore, no physics testing will be performed in the future. Therefore, the provisions of LCO 3.0.7 are no longer necessary. The NRC staff finds that the deletion of LCO 3.0.7 appropriately reflects the permanently shutdown and defueled status of the facility.

The NRC staff has reviewed the proposed changes to TS SR 3.0.2. The NRC staff agrees that the statements to be deleted are no longer necessary because the defueled TSs do not contain Frequencies and Completion Times of the type described in the statements being deleted. Therefore, the NRC staff finds the proposed changes acceptable.

Based on the above, the NRC staff finds that the licensee's proposed changes to LCO 3.0.1, LCO 3.0.2, LCO 3.0.3, LCO 3.0.4, LCO 3.0.5, LCO 3.0.6, LCO 3.0.7, SR 3.0.1, SR 3.0.2, and SR 3.0.4 are acceptable.

3.7.7 Section 3.1, Reactivity Control Systems

Section 3.1, "Reactivity Control Systems," of SONGS Units 2 and 3 TSs, contain LCOs, Actions, and SRs that provide for appropriate control of process variables, design features, or operating restrictions that are required to protect the integrity of a fission product barrier. The following TSs for SONGS Units 2 and 3 are being proposed for deletion.

TS 3.1.1, "SHUTDOWN MARGIN (SDM) – $T_{avg} > 200$ °F [degrees Fahrenheit]," specifies the requirements to provide sufficient reactivity margin to ensure that acceptable fuel design limits will not be exceeded for normal shut down and AOOs assuming the highest reactivity worth CEA remains fully withdrawn. TS 3.1.1 is applicable in "MODES 3 and 4."

TS 3.1.2, "SHUTDOWN MARGIN (SDM) - $T_{avg} \leq 200$ °F," specifies the requirements to provide sufficient reactivity margin to ensure that acceptable fuel design limits will not be exceeded for

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normal shut down and AOOs assuming the highest reactivity worth CEA remains fully withdrawn. TS 3.1.2 is applicable in "MODE 5."

TS 3.1.3, "Reactivity Balance," specifies the requirements for the comparison of the predicted versus measured core reactivity during power operation. The periodic confirmation of core reactivity is necessary to ensure that DBAs and transient safety analyses remain valid. TS 3.1.3 is applicable in "MODES 1 and 2."

TS 3.1.4, "Moderator Temperature Coefficient (MTC)," specifies the requirements to ensure that core overheating and overcooling accidents will not violate the accident analysis assumptions. TS 3.1.4 is applicable in "MODES 1 and 2 with k_{eff} [effective multiplication factor] ≥ 1.0."

TS 3.1.5, "Control Element Assembly (CEA) Alignment," specifies the limits on shutdown and regulating CEA alignments to ensure that the power distribution and reactivity limits defined by the design power peaking and SDM limits are preserved. TS 3.1.5 is applicable in "MODES 1 and 2."

TS 3.1.6, "Shutdown Control Element Assembly (CEA) Insertion Limits," specifies the limits on shutdown CEA insertion to ensure that a sufficient amount of negative reactivity is available to shut down the reactor and maintain the required shutdown margin following a reactor trip. TS 3.1.6 is applicable in "MODE 1, MODE 2 with any regulating CEA not fully inserted."

TS 3.1.7, "Regulating CEA Insertion Limits," specifies the limits on regulating CEA sequence and physical insertion for the function of preserving power distribution, ensuring that the SDM is maintained, ensuring that ejected CEA worth is maintained, and ensuring adequate negative reactivity insertion is available on trip. The overlap between regulating banks provides more uniform rates of reactivity insertion. TS 3.1.7 is applicable in "MODES 1 and 2."

TS 3.1.8, "Part Length Control Element Assembly (CEA) Insertion Limits," specifies the limits on part length CEA insertion for the function of preserving power distribution and ensuring that ejected CEA worth is maintained within limits. TS 3.1.8 is applicable in "MODE 1 > 20% RTP."

TS 3.1.9, "Boration Systems - Operating," establishes the requirements for borated water sources and flow paths to the RCS to ensure that sufficient borated water is available to maintain the reactor subcritical and provide makeup water to account for RCS shrinkage during cooldown to cold shutdown conditions. TS 3.1.9 is applicable in "MODES 1, 2, 3, and 4."

TS 3.1.10, "Boration Systems – Shutdown," establishes the requirements for borated water sources and flow paths to the RCS to ensure that sufficient borated water is available to maintain the reactor subcritical. TS 3.1.10 is applicable in "MODES 5 and 6."

TS 3.1.12, "Special Test Exception (STE) - Low Power Physics Testing," permits the relaxation of existing TS LCOs to allow the performance of PHYSICS TESTS. TS 3.1.12 is applicable in "MODES 2 and 3 during PHYSICS TESTS."

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TS 3.1.13, "Special Test Exception (STE) - At Power Physics Testing," permits relaxation of existing TS LCOs to allow the performance of PHYSICS TESTS. TS 3.1.13 is applicable in "MODE 1 during PHYSICS TESTS."

TS 3.1.14, "Special Test Exceptions (STE) - Reactivity Coefficient Testing," permits relaxation of existing TS LCOs to allow the performance of PHYSICS TESTS. TS 3.1.14 is applicable in "MODE 1."

The NRC staff has reviewed the licensee's proposed change to delete the reactivity control system TSs for SONGS Units 2 and 3, and has determined that these TSs are only needed to provide the LCOs and SRs necessary to maintain reactivity parameters of fuel loaded into a reactor vessel within the margins of conditions encountered during normal operations, anticipated occurrences, and for DBAs. The reactivity control systems TSs are only important for a reactor authorized to operate or retain irradiated fuel in the reactor vessel. However, because 10 CFR 50.82(a)(2) prohibits the licensee from operating the reactors or placing fuel in the reactor vessels at SONGS Units 2 and 3, there is no reactor core and the reactivity control systems are no longer relevant.

The NRC staff has also reviewed Section 3.1, the reactivity control systems TSs proposed for deletion (TS 3.1.1, TS 3.1.2, TS 3.1.3, TS 3.1.4, TS 3.1.5, TS 3.1.6, TS 3.1.7, TS 3.1.8, TS 3.1.9, TS 3.1.10, TS 3.1.12, TS 3.1.13, and TS 3.1.14), to ensure that these LCOs no longer satisfy the 10 CFR 50.36 criteria for inclusion in TSs, as described in Section 2.1 of this evaluation. These TSs indicate MODES for which the TSs are applicable. MODES, as defined in TSs, correspond to any one inclusive combination of core reactivity condition, power level, average reactor coolant temperature, and reactor vessel head closure bolt tensioning with fuel in the reactor vessel. The reference to MODES for permanently shutdown and defueled reactors, such as SONGS Units 2 and 3, has no meaning and is not relevant. Because SCE has submitted certifications pursuant to 10 CFR 50.82(a)(2), it is prohibited from operating the reactors or placing fuel in the reactor vessels and SONGS Units 2 and 3 are no longer in a configuration or a condition under which the TS MODES apply.

Therefore, the NRC staff finds that the licensee's proposed change to delete TS Section 3.1, Reactivity Control Systems, is acceptable.

3.7.8 Section 3.2, Power Distribution Limits

Section 3.2, "Power Distribution Limits," of the SONGS Units 2 and 3 TSs, contains LCOs, Required Actions, and SRs that provide for appropriate control of process variables, design features, or operating restrictions that are required to control power distribution in the reactor, and in turn, protect the integrity of a fission product barrier. The following TSs are being proposed for deletion.

TS 3.2.1, "Linear Heat Rate (LHR)," specifies the limits based on correlations between power peaking and certain measured variables used as inputs to the LHR and DNBR operating limits. The limitation on LHR ensures that in the event of a loss-of-coolant accident (LOCA) the peak

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temperature of the fuel cladding does not exceed 2200 °F. TS 3.2.1 is applicable in MODE 1 with THERMAL POWER > [greater than] 20% RTP.

TS 3.2.2, "Planar Radial Peaking Factors (F_{xy})," specifies the limits based on correlations between power peaking and certain measured variables used as inputs to the LHR and DNBR operating limits. Limiting of the calculated Planar Radial Peaking Factors to values equal to or greater than the measured Planar Radial Peaking Factors ensures that the calculated limits remain valid. TS 3.2.2 is applicable in "MODE 1 with THERMAL POWER > 20 % RTP."

TS 3.2.3, "AZIMUTHAL POWER TILT (Tq)," specifies the limits based on correlations between power peaking and certain measured variables used as inputs to the LHR and DNBR operating limits. The limitations on the Tq are provided to ensure that design operating margins are maintained. TS 3.2.3 is applicable in "MODE 1 with THERMAL POWER > 20% RTP."

TS 3.2.4, "Departure from Nucleate Boiling Ratio (DNBR)," specifies the limits based on correlations between power peaking and certain measured variables used as inputs to the LHR and DNBR operating limits. Operation of the core with a DNBR at, or above, this limit ensures that an acceptable minimum DNBR is maintained in the event of a loss of flow transient. TS 3.2.4 is applicable in "MODE 1 with THERMAL POWER > 20% RTP."

TS 3.2.5, "AXIAL SHAPE INDEX (ASI)," specifies the limits based on correlations between power peaking and certain measured variables used as inputs to the LHR and DNBR operating limits. The limitation on ASI ensures that the actual ASI value is maintained within the range of values used in the accident analysis. The ASI limits ensure that with T_q at its maximum upper limit, the DNBR does not drop below the DNBR safety limit for AOOs. TS 3.2.5 is applicable in "MODE 1 with THERMAL POWER > 20% RTP."

The NRC staff has reviewed the proposed deletion of the power distribution limits TSs and has determined that these TSs are only needed to provide the LCOs and SRs necessary to maintain reactor power and heat generation within the margins of conditions encountered during normal operation, anticipated occurrences, and for DBAs. The power distribution limits TSs are only important for a reactor authorized to operate. Because 10 CFR 50.82(a)(2) prohibits the licensee from operating the reactors or placing fuel in the reactor vessels at SONGS Units 2 and 3, there is no reactor core generating power and the power distribution limits are no longer relevant.

The NRC staff has also reviewed Section 3.2, power distribution limits TSs proposed for deletion (TS 3.2.1, TS 3.2.2, TS 3.2.3, TS 3.2.4, and TS 3.2.5), to ensure that these LCOs no longer satisfy the 10 CFR 50.36 criteria for inclusion in TSs, as described in Section 2.1 of this evaluation. These TSs indicate MODES for which the TSs are applicable. MODES, as defined in TSs, correspond to any one inclusive combination of core reactivity condition, power level, average reactor coolant temperature, and reactor vessel head closure bolt tensioning with fuel in the reactor vessel. The reference to MODES for permanently shutdown and defueled reactors, such as SONGS Unit 2 and Unit 3, has no meaning and is not relevant. Because SCE has submitted certifications pursuant to 10 CFR 50.82(a)(2), it is prohibited from operating the

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reactors or placing fuel in the reactor vessels and SONGS Unit 2 and Unit 3 are no longer in a configuration or a condition under which the TS MODES apply.

Therefore, the NRC staff finds that the licensee's proposed change to delete TS Section 3.2, Power Distribution Limits, is acceptable.

3.7.9 Section 3.3, Instrumentation

Section 3.3, "Instrumentation," of the SONGS Units 2 and 3 TSs, contains the LCOs, Actions, and SRs that provide for appropriate functional capability of sensing and control instrumentation required for the appropriate functional capability of plant equipment, and control of process variables, design features, or operating restrictions required for safe operation of the facility. The following TSs are being proposed for deletion.

TS 3.3.1, "Reactor Protective System (RPS) Instrumentation-Operating," specifies the requirements for the RPS instrumentation system to maintain the SLs during AOOs, and mitigates the consequences of DBAs in MODE 1 and MODE 2 (startup).

TS 3.3.2, "Reactor Protective System (RPS) Instrumentation – Shutdown," specifies the requirements for the RPS instrumentation system to maintain the SLs during all AOOs, and mitigates the consequences of DBAs in MODE 3 (hot standby), MODE 4 (hot shutdown), and MODE 5, when the reactor trip circuit breakers (RTCBs) are closed and the CEA drive system is capable of CEA withdrawal.

TS 3.3.3, Control Element Assembly Calculators (CEACs)," specifies the requirements to ensure the core protection calculators (CPCs) are either informed of individual CEA position within each subgroup, using one or other CEACs, or that appropriate conservatism is included in the CPC calculations to account for anticipated CEA deviations in MODES 1 and 2.

TS 3.3.4, "Reactor Protective System (RPS) Logic and Trip Initiation," specifies the requirements for RPS matrix logic, RPS initiation logic, RTBCs, and manual trip channels to effect automatic trip signals received from RPS instruments and to provide a means to manually trip the reactor in MODES 1 and 2, and in MODES 3, 4, and 5, when the RTCBs are closed and the CEA drive system is capable of CEA withdrawal.

TS 3.3.5, "Engineered Safety Features Actuation System (EFSAS) Instrumentation," specifies the requirements for the ESFAS Instrumentation to ensure ESFAS initiates necessary safety systems, based on the values of selected unit parameters, to protect against violating core design limits and the RCPB during AOOs and ensures acceptable consequences during accidents.

TS 3.3.6, "Engineered Safety Features Actuation System (EFSAS) Logic and Manual Trip," specifies the requirements for ESFAS Matrix Logic, ESFAS Initiation Logic, and Manual Trip channels to effect automatic ESFAS initiation received from ESFAS instruments and to provide a means to manually actuate an ESF system in MODES 1, 2, 3, and 4.

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TS 3.3.11, "Post Accident Monitoring Instrumentation (PAMI)," is applicable in MODES 1, 2, and 3, and provides the operability requirements for accident monitoring instruments, which provides information required by the control room operators. The operability of the PAMI ensures there is sufficient information available on selected unit parameters to monitor and assess unit status following an accident.

TS 3.3.12, "Remote Shutdown System," is applicable in MODES 1, 2, and 3, and provides the operability requirements for instrumentation and controls necessary to place and maintain the unit in MODE 3 from a location other than the control room.

TS 3.3.13, "Source Range Monitoring Channels," is applicable in MODES 3, 4, and 5, and provides the operability requirements for source range monitoring instrumentation for indication, alarms, and reactor trips.

The NRC staff has reviewed the proposed deletion of the instrumentation TSs above. TS 3.3.1, TS 3.3.2, TS 3.3.3 and TS 3.3.4, are only necessary to maintain the ability of the RPS to automatically initiate a reactor scram to preserve the integrity of the fuel cladding, preserve the integrity of the primary system barrier, and minimize the energy which must be absorbed, and prevent criticality following a LOCA. TS 3.3.5 and TS 3.3.6 only concern instrumentation designed to mitigate accidents related to reactor operation. TS 3.3.11 and TS 3.3.12 only concern instrumentation to ensure there is sufficient information available on selected unit parameters to monitor and assess unit status following an accident, and allow operators to take manual actions specified in the emergency operating procedures or to remotely shut down the reactor from a location other than the control room. TS 3.3.13 requires monitoring source range count rate level and detects a loss of SDM caused by a boron dilution event (detected as an increase in neutron flux). None of the instrumentation addressed by these TSs is needed by a reactor that has permanently shut down and defueled in accordance with 10 CFR 50.82(a)(2).

The NRC staff also has reviewed the above Section 3.3 instrumentation TSs that are proposed for deletion (TS 3.3.1, TS 3.3.2, TS 3.3.3, TS 3.3.4, TS 3.3.5, TS 3.3.6, TS 3.3.11, TS 3.3.12, and TS 3.3.13), to ensure that these LCOs no longer satisfy the 10 CFR 50.36 criteria for inclusion in TSs, as described in Section 2.1 of this evaluation. These TSs indicate MODES for which the TSs are applicable. MODES, as defined in TSs, correspond to any one inclusive combination of core reactivity condition, power level, average reactor coolant temperature, and reactor vessel head closure bolt tensioning with fuel in the reactor vessel. The reference to MODES for permanently shutdown and defueled reactors, such as SONGS Unit 2 and Unit 3, has no meaning and is not relevant. Because SCE has submitted certifications pursuant to 10 CFR 50.82(a)(2), it is prohibited from operating the reactors or placing fuel in the reactor vessels and SONGS Units 2 and 3 are no longer in a configuration or a condition under which the TS MODES apply.

Therefore, the NRC staff finds that the licensee's proposed change to delete Section 3.3, instrumentation TS 3.3.1, TS 3.3.2, TS 3.3.3, TS 3.3.4, TS 3.3.5, TS 3.3.6, TS 3.3.11, TS 3.3.12, and TS 3.3.13 is acceptable.

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TS 3.3.7, "Diesel Generator (DG) – Undervoltage Start," specifies the loss of voltage start (LOVS) instrumentation functions be operable in MODES 1, 2, 3, 4, and when the associated DG is required to be operable by LCO 3.8.2, AC Sources – Shutdown. TS 3.3.7 is proposed for deletion by the licensee.

TS 3.3.7 provides the LCO and SRs to ensure availability of backup safety-related alternating current (AC) power to the SSCs used to prevent or mitigate postulated accidents resulting in an uncontrolled release of radioactivity of DBAs as analyzed in the SONGS Unit 2 and 3 UFSARs. This TS LCO ensures the operability of instrumentation designed to detect an undervoltage on the safety-related AC electrical busses upon a loss of offsite power and start the emergency diesel generators (EDGs) to supply backup power to the AC busses. Since SONGS Units 2 and 3 are permanently shut down and defueled, SCE has analyzed the remaining DBAs at SONGS Units 2 and 3, and given the significant fuel decay period, found that the radiological consequences will not exceed the EPA's PAGs at the site boundary. The licensee's analysis also demonstrates that the dose consequences, within the CR, of any DBAs are acceptable without relying on SSCs remaining functional for accident mitigation except the passive fuel storage pool structure which will be maintained as a TS for SONGS.) SCE calculated the DBA radiological consequences assuming no credit for control room isolation or recirculation filtration and no credit for any accident mitigation by the auxiliary building ventilation system. Calculated doses at the EAB, LPZ, and CR are within 10 CFR 50.67 and RG 1.183 dose limits. Since the bounding accident analysis for the permanently defueled condition assumed no credit for control room post-accident recirculation system emergency ventilation or filtration for mitigation of radiological releases, the CREACUS is not required (see evaluation of TS 3.7.11 in Section 3.7.13 of this SE). Because CREACUS is not required to function to mitigate a DBA, backup electrical power required to support operation of CREACUS upon a loss of offsite power is unnecessary. Therefore the EDG loss of offsite power start instrumentation is also unnecessary for all postulated DBAs.

The NRC staff determined in Section 3.1 through 3.6 of this SE that with SONGS Units 2 and 3 permanently shut down and defueled and the irradiated fuel having decayed for a significant period, CREACUS is no longer needed or credited in the primary success path of a safety sequence analysis related the remaining DBAs at SONGS Units 2 and 3. Consequently, neither primary nor backup power to support operation of CREACUS is needed. Therefore, actuation instrumentation to start the backup power EDGs is no longer required to satisfy TS Criterion 3 for inclusion in TSs as a support or actuation system that is necessary for items in the primary success path to successfully function. The NRC staff has confirmed that there are no other DBAs analyzed in the SONGS Units 2 and 3 UFSAR that rely on this instrumentation system. Based on the above, the NRC staff finds that the licensee's proposed change to delete TS 3.3.7 is acceptable.

TS 3.3.8, "Containment Purge Isolation Signal (CPIS)," specifies the requirements for instrumentation designed to close the containment purge isolation valves upon a detection of high gaseous radiation in containment. This action isolates the containment atmosphere from the environment to minimize releases of radioactivity in the event of an accident. TS 3.3.8 is applicable in MODES 1, 2, 3, and 4, during core alterations, and during movement of fuel assemblies within containment. TS 3.3.8 is proposed for deletion by the licensee.

This TS indicates MODES for which the TS is applicable. MODES, as defined in TSs, correspond to any one inclusive combination of core reactivity condition, power level, average reactor coolant temperature, and reactor vessel head closure bolt tensioning with fuel in the reactor vessel. The reference to MODES for permanently shutdown and defueled reactors, such as SONGS Unit 2 and Unit 3, has no meaning and is not relevant. Because SCE has submitted certifications pursuant to 10 CFR 50.82(a)(2), it is prohibited from operating the reactors or placing fuel in the reactor vessels and SONGS Units 2 and 3 are no longer in a configuration or a condition under which the TS MODES apply. The staff also reviewed the non-MODE dependent applicability during core alterations, and during movement of fuel assemblies in containment. Because 10 CFR 50.82(a)(2) prohibits operation of the plant or placing fuel in the reactor vessel, the prohibition on placing fuel in the reactor vessel, it also precludes core alterations and the movement of fuel assemblies within containment.

The NRC staff also evaluated the proposed deletion of TS 3.3.8, to ensure that the LCO no longer satisfies the 10 CFR 50.36 criteria for inclusion in TSs, as described in Section 2.1 of this evaluation. The staff has determined that TS 3.3.8 only addresses specific plant systems, control of process variables, design features, or operating restrictions associated with the containment and are no longer needed or credited in the primary success path of a safety sequence analysis related the remaining DBAs at SONGS Units 2 and 3. Based on the above, the NRC staff finds that the licensee's proposed change to delete TS 3.3.8 is acceptable.

TS 3.3.9, "Control Room Isolation Signal (CRIS)," specifies the requirements to ensure instrumentation (actuation logic, manual trip, and gaseous radiation monitors) necessary to initiate CREACUS is operable. The CRIS terminates the normal supply of outside air to the CR and initiates actuation of the CREACUS to minimize operator radiation exposure. The radiation monitor actuation of the CREACUS in MODES 5 and 6 and during movement of fuel assemblies is the primary means to ensure control room habitability in the event of an FHA. TS 3.3.9 is applicable in MODES 1, 2, 3, 4, 5, 6, and during movement of fuel assemblies within containment and in the fuel storage pool. TS 3.3.9 is proposed for deletion by the licensee.

This TS indicates MODES for which the TS is applicable. MODES, as defined in TSs, correspond to any one inclusive combination of core reactivity condition, power level, average reactor coolant temperature, and reactor vessel head closure bolt tensioning with fuel in the reactor vessel. The reference to MODES for permanently shutdown and defueled reactors, such as SONGS Units 2 and 3, has no meaning and is not relevant. Because SCE has submitted certifications pursuant to 10 CFR 50.82(a)(2), it is prohibited from operating the reactors or placing fuel in the reactor vessels and SONGS Units 2 and 3 are no longer in a configuration or a condition under which the TS MODES apply.

The NRC staff evaluated the proposed deletion of TS 3.3.9 to ensure that the LCO no longer satisfies the 10 CFR 50.36 criteria for inclusion in TSs, as described in Section 2.1 of this evaluation. The NRC staff also reviewed the non-MODE dependent applicability during movement of fuel assemblies within containment and in the fuel storage pool. As detailed in Sections 3.1 through 3.6 of this SE, the remaining accident analyses applicable to the permanently shutdown and defueled reactors of SONGS Units 2 and 3 show that the dose

consequences within the control room are acceptable. Furthermore, the analyses do not rely on the functioning of any SSCs to mitigate the postulated DBAs with the exception of the passive fuel storage pool structure.

The NRC staff evaluated the remaining DBAs that credited the CREACUS, and its support systems, that were previously relied upon to mitigate the CR, EAB or LPZ dose consequences during reactor operation. This includes the FHIS, the FHB PACU filtration system, and the CRIS. As discussed in the basis for deleting CREACUS TS 3.7.11, (see Section 3.7.13 of this SE), the CRIS is no longer required for providing airborne radiological protection for the control room operators in the event of a DBA. Since TS 3.3.9 exists solely to support CREACUS Operability, the elimination of the need for the CREACUS also obviates the need for its support systems. The deletion of the CREACUS TS 3.7.11 eliminates the need for the CRIS TS 3.3.9. Therefore, the NRC staff finds the CRIS isolation signal is no longer required and that the licensee's proposed change to delete TS 3.3.9 is acceptable.

3.7.10 Section 3.4, Reactor Coolant System

The RCS TSs of Section 3.4, "Reactor Coolant System (RCS)," for SONGS Units 2 and 3 contain the LCOs, Actions, and SRs that provides for appropriate control of process variables, design features, or operating restrictions needed for appropriate functional capability of RCS equipment required for safe operation of the facility. The following TSs are being proposed for deletion.

TS 3.4.1, "RCS DNB [Departure from Nucleate Boiling] (Pressure, Temperature, and Flow) Limits," specifies the process variables requirements for maintaining RCS pressure, temperature, and flow rate within limits assumed in the safety analyses. The limits placed on RCS pressure, temperature, and flow rate ensure that the minimum DNBR will be met for each of the analyzed transients. TS 3.4.1 is applicable in MODE 1.

TS 3.4.2, "RCS Minimum Temperature for Criticality," specifies the requirements for RCS loop cold leg temperature (T_c) before the reactor can be made critical and while the reactor is critical. Compliance with the LCO ensures that the reactor will not be made or maintained critical ($k_{eff} > 1.0$) outside a temperature operating range of 522 °F to 558 °F, and to prevent operation in an unanalyzed condition. TS 3.4.2 is applicable in "MODE 1, THERMAL POWER ≤ 30% RTP and $T_c < 535$ °F, and in MODE 2, $K_{eff} \geq 1.0$ and $T_c < 535$ °F."

TS 3.4.3, "RCS Pressure and Temperature (P/T) Limits," specifies that the RCS pressure, RCS temperature and RCS heatup and cooldown rates shall be maintained within the limits as specified in the Pressure - Temperature Limits Report (PTLR). The LCO establishes operating limits that provide a margin to brittle failure of the reactor vessel and piping of the RCPB. TS 3.4.3 is applicable at all times. The purpose for TS LCO 3.4.3 during normal operation of the RCS is to avoid encountering pressure, temperature, and temperature rate of change conditions that might cause undetected flaws to propagate and cause nonductile failure of the RCPB, an unanalyzed condition. The RCS P/T limits in LCO 3.4.3 provide a definition of acceptable operation for prevention of nonductile failure in accordance with 10 CFR Part 50, Appendix G. Although the P/T limits were developed to provide guidance for operation during heatup, or

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cooldown, or inservice leak and hydrostatic testing, the applicability of these limits is at all times in keeping with the concern for nonductile failure.

TS 3.4.3.1, "Pressurizer Heatup and Cooldown Limits," requires that the pressurizer heatup and cooldown rates shall be maintained within the specified limits. The pressurizer is designed to withstand the effects of cyclic loads due to system temperature and pressure changes. These cyclic loads are introduced by normal load transients, reactor trips, and startup and shutdown operations. During startup and shutdown, the rates of temperature and pressure changes are limited so that the maximum specified heatup and cooldown rates are consistent with the design assumptions and satisfy the stress limits for cyclic operation. Therefore, TS 3.4.3.1 is applicable at all times.

TS 3.4.4, "RCS Loops - MODES 1 and 2," specifies the requirements to ensure heat removal capability of the RCS loops with the reactor in MODES 1 and 2. The primary function of the RCS is removal of the heat generated in the fuel due to the fission process, and transfer of this heat, via the steam generators (SGs), to the secondary plant. The important aspect for this LCO is the reactor coolant forced flow rate, which is represented by the number of RCS loops in service. TS 3.4.4 is applicable in MODES 1 and 2.

TS 3.4.5, "RCS Loops - MODE 3," specifies the requirements to ensure heat removal capability of the RCS loops with the reactor in MODE 3. In MODE 3, the primary function of the reactor coolant is removal of decay heat and transfer of this heat, via the SGs, to the secondary plant fluid. TS 3.4.5 is applicable in MODE 3.

TS 3.4.6, "RCS Loops - MODE 4," specifies the requirements to ensure heat removal capability of the RCS loops with the reactor in MODE 4. In MODE 4, the primary function of the reactor coolant is the removal of decay heat and the transfer of this heat to the SGs or shutdown cooling (SDC) heat exchangers. TS 3.4.6 is applicable in MODE 4.

TS 3.4.7, "RCS Loops - MODE 5, Loops Filled," specifies the requirements to ensure heat removal capability of the RCS loops with the reactor in MODE 5 with the RCS loops filled with coolant. In MODE 5 with the RCS loops filled, the primary function of the reactor coolant is the removal of decay heat and transfer of this heat to the SGs or SDC heat exchangers. While the principal means for decay heat removal is via the SDC heat exchangers, the SGs are specified as a backup means for redundancy. TS 3.4.7 is applicable in MODE 5 with the RCS loops filled.

TS 3.4.8, "RCS Loops - MODE 5, Loops Not Filled," specifies the requirements to ensure heat removal capability of the RCS loops with the reactor in MODE 5 with the RCS loops not filled with reactor coolant. In MODE 5 with the RCS loops not filled, the primary function of the reactor coolant is the removal of decay heat and transfer of this heat to the SDC heat exchangers. The SGs are not available as a heat sink when the loops are not filled. TS 3.4.8 is applicable in MODE 5 with the RCS loops not filled.

TS 3.4.9, "Pressurizer," specifies the OPERABILITY requirements for the RCS pressurizer. The pressurizer provides a point in the RCS where liquid and vapor are maintained in equilibrium

under saturated conditions for pressure control purposes to prevent bulk boiling in the remainder of the RCS. In MODES 1, 2, and 3, the LCO requirement for a steam bubble is reflected implicitly in the accident analyses. All analyses performed from a critical reactor condition assume the existence of a steam bubble and saturated conditions in the pressurizer. TS 3.4.9 is applicable in MODES 1, 2, and 3.

TS 3.4.10, "Pressurizer Safety Valves," specifies the OPERABILITY and lift setpoint parameters for the pressurizer safety valves. The pressurizer safety valves provide, in conjunction with the reactor protection system, overpressure protection for the RCS. The pressurizer safety valves are designed to prevent the RCS from exceeding the system safety limit of 2750 pounds per square inch absolute (psia) in MODES 1, 2, and 3. In MODES 4, 5, and MODE 6 with the reactor vessel head on, overpressure protection is provided by operating procedures and by meeting the requirements of LCO 3.4.12, Low Temperature Overpressure Protection (LTOP) System. TS 3.4.10 is applicable in MODES 1, 2, and 3.

TS 3.4.12.1, "Low Temperature Overpressure Protection (LTOP) System, RCS Temperature ≤ PTLR Limit," specifies the requirements for controlling RCS pressure at low temperatures so the integrity of the RCPB is not compromised by violating the P/T limits of 10 CFR Part 50, Appendix G. TS LCO 3.4.12.1 provides RCS overpressure protection by minimizing coolant input capability and having adequate pressure relief capacity. In MODES 1, 2, and 3, the pressurizer safety valves will prevent RCS pressure from exceeding limits. In MODE 4 when the temperature of any RCS cold leg is less than or equal to the enable temperature specified in the PTLR, MODE 5, and MODE 6 when the reactor vessel head is on and the RCS is not vented, overpressure prevention falls to the OPERABLE SDC system relief valve or to a depressurized RCS and a sufficient sized RCS vent. When the reactor vessel head is off, overpressurization cannot occur.

TS 3.4.12.2, "Low Temperature Overpressure Protection (LTOP) System, RCS Temperature ≥ PTLR Limit," specifies requirements for controlling RCS pressure at low temperatures so the integrity of the RCPB is not compromised by violating the P/T limits of 10 CFR Part 50, Appendix G. TS LCO 3.4.12.2 provides RCS overpressure protection by having adequate pressure relief capacity. In MODES 1, 2, and 3 the pressurizer safety valves will prevent RCS pressure from exceeding limits. In MODE 4 when the temperature of all RCS cold legs are greater than the enable temperature specified in the PTLR, overpressure prevention falls to the OPERABLE SDC system relief valve or to an OPERABLE pressurizer code safety valve.

TS 3.4.13, "RCS Operational LEAKAGE," specifies the process variable limits and operating restrictions for RCS pressure boundary leakage, unidentified RCS leakage, identified RCS leakage, and primary to secondary leakage. RCS leakage is indicative of material deterioration, possibly of the RCS pressure boundary, which can affect the probability of a design basis event. The primary to secondary leakage limit is based on operating experience with SG tube degradation mechanisms that result in tube leakage. The operational leakage rate criterion in conjunction with the implementation of the Steam Generator Program is an effective measure for minimizing the frequency of steam generator tube ruptures (SGTRs). TS 3.4.13 is applicable in MODES 1, 2, 3, and 4. In MODES 5 and 6, leakage limits are not required because the

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reactor coolant pressure is far lower, resulting in lower stresses and reduced potential for leakage.

TS 3.4.14, "RCS Pressure Isolation Valve (PIV) Leakage," specifies the process variable limits and operating restrictions for RCS PIV leakage. The regulations in 10 CFR 50.2, 10 CFR 50.55a(c), and 10 CFR Part 50, Appendix A, GDC 55, discuss RCPB valves, which are normally closed valves in series within the RCPB that separate the high pressure RCS from an attached low pressure system. Although this specification provides a limit on allowable PIV leakage rate, its main purpose is to prevent overpressure failure of the low pressure portions of connecting systems (intersystem LOCA). PIVs are provided to isolate the RCS from the following typically connected systems: SDC system; safety injection system; and the chemical and volume control system. The leakage limit is an indication that the PIVs between the RCS and the connecting systems are degraded or degrading. TS 3.4.14 is applicable in MODES 1, 2, 3, and 4. In MODES 5 and 6, leakage limits are not provided because the lower reactor coolant pressure results in a reduced potential for PIV leakage outside the containment.

TS 3.4.15, "RCS Leakage Detection Instrumentation," specifies the OPERABILITY requirements for RCS leakage detection instrumentation. Leakage detection systems are provided to detect significant RCPB degradation as soon after occurrence as practical to minimize the potential for propagation to a gross failure. Thus, they provide an early indication or warning signal to permit proper evaluation of RCS leakage into the containment area. TS LCO 3.4.15 requires instruments of diverse monitoring principles to be OPERABLE to provide a high degree of confidence that extremely small leaks are detected in time to allow actions to place the plant in a safe condition, when RCS leakage indicates possible RCPB degradation. TS 3.4.15 is applicable in MODES 1, 2, 3, and 4.

TS 3.4.16, "RCS Specific Activity," specifies the process variable limits and operating restrictions for Dose Equivalent 1-131 and gross specific activity. The TS LCO limits on the specific activity of the reactor coolant ensure that the resulting offsite doses meet the appropriate RG 1.183 acceptance criteria following a SGTR accident. TS 3.4.16 is applicable in MODES 1, 2, and MODE 3 with RCS average temperature ≥ 500 degrees F.

TS 3.4.17, "Steam Generator (SG) Tube Integrity," specifies the requirements to ensure the RCPB integrity function of the SG. The SGTR accident is the limiting design basis event for SG tubes and avoiding an SGTR is the basis for this specification. TS 3.4.17 is applicable in MODES 1, 2, 3, and 4.

The licensee proposed to delete all of Section 3.4 of the SONGS Units 2 and 3, RCS TSs, since all except TS 3.4.3, are only applicable to operating reactor MODES and do not apply to a permanently shutdown and defueled reactor. The NRC staff has reviewed these proposed changes and has determined that these TSs are only needed to provide the LCOs and SRs necessary to maintain functionality and integrity of the RCS pressure boundary. These TSs contain requirements for various RCS parameters such as: thermal limitations for heatup and cooldown rates during plant operation in order to operate within the analyzed requirements for stress intensity and fatigue limits for the reactor vessel; pressurization, which established and maintained an equilibrium under saturated conditions for pressure control to prevent bulk boiling

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in the remainder of the RCS; coolant chemistry, which included limits on RCS activity to limit potential offsite doses due to postulated events and limits on RCS conductivity, chlorides, and pH to prevent stress-corrosion cracking; coolant leakage, which established primary system leakage limits to allow prompt identification and isolation of leaks before the integrity of the RCS pressure boundary was impaired; safety and relief valves, which specifies operability requirements for the safety and relief valves designed to prevent overpressurization of, and damage to, the primary system boundary; and structural integrity, which addresses the inservice inspection requirements of the primary system boundary components. All of these TSs are related to assuring the integrity of the RCS pressure boundary. The RCS TSs are only important for a reactor authorized to operate or retain irradiated fuel in the reactor vessel. However, because 10 CFR 50.82(a)(2) prohibits the licensee from operating the reactors or placing fuel in the reactor vessels at SONGS Units 2 and 3, the RCS is no longer functional or used in any capacity.

Regarding the applicability of TS 3.4.3 and TS 3.4.3.1 at all times, the NRC staff notes that the RCS and pressurizer are drained and vented, to the extent possible, and consequently there is no longer any concern about exceeding the RCS and pressurizer P/T or cyclic limits. The requirements of 10 CFR Part 50, Appendix G, no longer apply to a permanently shutdown and defueled reactor because the RCPB will no longer be used as a fission product barrier. Therefore, TS 3.4.3 is no longer needed and may be deleted. Similarly, operating the unit within the fatigue analysis performed in accordance with the ASME Code Section III requirements no longer applies. Therefore, TS 3.4.3.1 is no longer needed and may also be deleted.

The NRC staff has also reviewed the RCS TSs proposed for deletion (TS 3.4.1, TS 3.4.2, TS 3.4.3, TS 3.4.3.1, TS 3.4.4, TS 3.4.5, TS 3.4.6, TS 3.4.7, TS 3.4.8, TS 3.4.9, TS 3.4.10, TS 3.4.12.1, TS 3.4.12.2, TS 3.4.13, TS 3.4.14, TS 3.4.15, TS 3.4.16, and TS 3.4.17), to ensure that these LCOs no longer satisfy the 10 CFR 50.36 criteria for inclusion in TSs, as described in Section 2.1 of this SE. The staff notes that these TSs indicate MODES for which these TSs are applicable. MODES, as defined in TSs, correspond to any one inclusive combination of core reactivity condition, power level, average reactor coolant temperature, and reactor vessel head closure bolt tensioning with fuel in the reactor vessel. The reference to MODES for a permanently shutdown and defueled reactor, such as SONGS Units 2 and 3, has no meaning and is not relevant. Because SCE has submitted certifications pursuant to 10 CFR 50.82(a)(2) for SONGS Units 2 and 3, it is prohibited from operating the reactors or placing fuel in the reactor vessels and, therefore, SONGS Units 2 and 3 are no longer in a configuration or a condition under which the TS MODES apply. Furthermore, because irradiated fuel has been permanently removed from the reactor pressure vessels, the RCS is no longer relevant as a fission product barrier.

Therefore, the NRC staff finds that the licensee's proposed change to delete TS Section 3.4, Reactor Coolant System, is acceptable.

3.7.11 Section 3.5, Emergency Core Cooling Systems (ECCS)

Section 3.5 of the SONGS Units 2 and 3 TSs, "Emergency Core Cooling Systems (ECCS)," contains LCOs that provide for appropriate functional capability of ECCS equipment required for

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mitigation of DBAs or transients so as to protect the integrity of a fission product barrier. The following TSs are being proposed for deletion.

TS 3.5.1, "Safety Injection Tanks (SITs)," specifies the requirements for the SITs to ensure they are capable of supplying water to the reactor vessel during the blowdown phase of a LOCA, to provide inventory to help accomplish the refill phase that follows thereafter, and to provide RCS makeup for a small-break LOCA. TS 3.5.1 is applicable in MODES 1 and 2, and in MODE 3 with pressurizer pressure ≥ 715 psia.

TS 3.5.2, "ECCS - Operating," specifies the requirements for the ECCS trains so as to provide core cooling and negative reactivity to ensure that the reactor core is protected after a LOCA, CEA ejection accident, loss of secondary coolant accident (including uncontrolled steam release), and SGTR. The ECCS consists of the high pressure safety injection (HPSI) and the low pressure safety injection (LPSI) subsystems. TS 3.5.2 is applicable in MODES 1 and 2, and MODE 3 with pressurizer pressure ≥ 400 psia.

TS 3.5.3, "ECCS - Shutdown," specifies the requirements for ECCS with the reactor in MODE 3 with pressurizer pressure < 400 psia, and in MODE 4. In these MODES, an ECCS train is composed of a single HPSI subsystem. One OPERABLE ECCS train is acceptable without a single failure consideration, based on the stable reactivity condition of the reactor and the limited core cooling requirements.

TS 3.5.4, "Refueling Water Storage Tank (RWST)," specifies the requirements for RWST OPERABILITY. During accident conditions, the RWST provides a source of borated water to the ECCS and containment spray system pumps. As such, it provides containment cooling and depressurization, core cooling, and replacement inventory and is a source of negative reactivity for reactor shutdown. TS 3.5.4 is applicable in MODES 1, 2, 3, and 4 because RWST OPERABILITY requirements are dictated by ECCS and containment spray system OPERABILITY requirements. Since both the ECCS and the containment spray system must be OPERABLE in MODES 1, 2, 3, and 4, the RWST must also be OPERABLE to support their operation.

TS 3.5.5, "Trisodium Phosphate (TSP) Dodecahydrate," specifies the requirements for TSP crystals to be placed in baskets on the floor of the containment building to ensure that iodine, which may be dissolved in the recirculated reactor cooling water following a LOCA, remains in solution. TSP also helps inhibit stress corrosion cracking (SCC) of austenitic stainless steel components in containment during the recirculation phase following an accident. TS 3.5.5 is applicable in MODES 1, 2, and 3, when the RCS is at elevated temperature and pressure, providing an energy potential for a LOCA.

The NRC staff has reviewed the proposed changes to the ECCS TSs and has determined that these TSs are only needed to provide the LCOs and SRs necessary to maintain functionality of the systems that provide emergency cooling to the reactor core and assure the appropriate functional capability ECCS required for mitigation of DBAs when the reactor is in MODES 1 through 4. These TSs includes multiple LCOs addressing the SITs, HPSI and LPSI subsystems; part of the ECCS, designed to provide adequate emergency cooling capability to

the reactor in the event of a LOCA; the RWST, designed to supply borated water to the ECCS during accident conditions; and the TSP baskets to help retain iodine in solution. All of these TSs are related to provide cooling for a reactor vessel core. Since SONGS Units 2 and 3 are permanently shut down and defueled, there are no accidents of any kind that would require emergency core cooling and the accidents that these systems and components were designed to mitigate are no longer possible.

The NRC staff also reviewed the ECCS TSs proposed for deletion (TS 3.5.1, TS 3.5.2, TS 3.5.3, TS 3.5.4, and TS 3.5.5), to ensure that these LCOs no longer satisfy the 10 CFR 50.36 criteria for inclusion in TSs, as described in Section 2.1 of this evaluation. The staff notes that these TSs indicate MODES for which the TSs are applicable. MODES, as defined in TSs, correspond to any one inclusive combination of core reactivity condition, power level, average reactor coolant temperature, and reactor vessel head closure bolt tensioning with fuel in the reactor vessel. The reference to MODES for a permanently shutdown and defueled reactor, such as SONGS Units 2 and 3, has no meaning and is not relevant. Because SCE has submitted certifications pursuant to 10 CFR 50.82(a)(2), it is prohibited from operating the reactors or placing fuel in the reactor vessels, and SONGS Units 2 and 3 are no longer in a configuration or a condition under which the TS MODES apply.

Therefore, the NRC staff finds that the licensee's proposed change to delete TS Section 3.5, Emergency Core Cooling Systems, is acceptable.

3.7.12 Section 3.6, Containment Systems

Section 3.6 of SONGS Units 2 and 3 TSs, "Containment Systems," contains the LCOs, Actions, and SRs that provide for appropriate control of process variables, design features, or operating restrictions required to protect the integrity of the containment as a fission product barrier; and appropriate functional capability of ESF equipment required for mitigation of DBAs or transients so as to protect the integrity of containment. The following TSs are being proposed for deletion from this section.

TS 3.6.1, "Containment," specifies the requirements for the containment to ensure it is capable of withstanding the pressures and temperatures of the limiting DBA without exceeding the design leakage rate. The containment steel liner and its penetrations establish the leakage limiting boundary of the containment. This TS provides the operating restrictions required to protect the integrity of the containment as a fission product barrier and limits the leakage of fission product radioactivity from the containment to the environment. TS 3.6.1 is applicable in MODES 1, 2, 3, and 4.

TS 3.6.2, "Containment Air Locks," specifies the requirements for the structural integrity and leak tightness of the containment air locks. As part of the containment pressure boundary, the air lock safety function is related to control of the containment leakage rate resulting from a DBA. Thus, each containment air lock's structural integrity and leak tightness is essential to the successful mitigation of such an event. TS 3.6.2 is applicable in MODES 1, 2, 3, and 4.

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TS 3.6.3, "Containment Isolation Valves," specifies the requirements for the isolation capability of the containment via the containment isolation valves. Containment isolation valves form a part of the containment boundary and their OPERABILITY supports leak tightness of the containment. TS 3.6.3 is applicable in MODES 1, 2, 3, and 4.

TS 3.6.4, "Containment Pressure," specifies the limitations on internal containment pressure. Containment internal pressure is an initial condition used in the DBA analyses to establish the maximum peak containment internal pressure. Maintaining containment pressure at less than or equal to the LCO upper pressure limit ensures that, in the event of a DBA, the resultant peak containment accident pressure will remain below the maximum allowed containment internal pressure. Maintaining containment pressure at greater than or equal to the LCO lower pressure limit ensures that the containment will not exceed the design negative differential pressure following the accidental actuation of the Containment Spray System. TS 3.6.4 is applicable in MODES 1, 2, 3, and 4.

TS 3.6.5, "Containment Air Temperature," specifies the limitations on containment average air temperature. Containment average air temperature is an initial condition used in the DBA analyses that establishes the containment environmental qualification operating envelope for both pressure and temperature. During a DBA, with an initial containment average air temperature less than or equal to the LCO temperature limit, the resultant accident temperature profile assures that the containment structural temperature is maintained below its design temperature and that required safety-related equipment will continue to perform its function. TS 3.6.5 is applicable in MODES 1, 2, 3, and 4.

TS 3.6.6.1, "Containment Spray and Cooling Systems," specifies the operability requirements for containment atmosphere cooling to limit post-accident pressure and temperature in containment to less than the design values. Reduction of containment pressure and the iodine removal capability of the spray reduce the release of fission product radioactivity from containment to the environment, in the event of a DBA, to within limits. The containment spray system consists of two separate trains. Each train includes a containment spray pump, spray headers, valves and piping. The RWST supplies borated water to the containment spray system during the injection phase of operation. In the recirculation mode of operation, containment spray pump suction is transferred from the RWST to the containment sump. Two trains of containment cooling, each of sufficient capacity to supply 50 percent of the design cooling requirement, are provided. Two trains with two fan units each are supplied with cooling water from the component cooling water system. All four fans are required to furnish the design cooling capacity. Air is drawn into the coolers through the fans and discharged to the SG compartments and pressurizer compartment. TS 3.6.6.1 is applicable in MODES 1, 2, and 3.

TS 3.6.6.2, "Containment Cooling Systems," specifies the operability requirements for containment atmosphere cooling to limit post-accident pressure and temperature in containment to less than the design values. Reduction of containment pressure reduces the release of fission product radioactivity from containment to the environment, in the event of a DBA, to within limits. Two trains of containment cooling, each of sufficient capacity to supply 50 percent of the design cooling requirement, are provided. Two trains with two fan units each are supplied with cooling water from the component cooling water system. All four fans are required to

furnish the design cooling capacity. Air is drawn into the coolers through the fans and discharged to the SG compartments and pressurizer compartment. TS 3.6.6.2 is applicable in MODE 4, when a DBA could cause a release of radioactive material to containment and an increase in containment pressure and temperature, requiring the operation of the containment cooling trains.

TS 3.6.8, "Containment Dome Air Circulators," specifies the requirements for the containment dome air circulators to reduce the potential for breach of the containment due to a hydrogen oxygen reaction. The dome air circulators accelerate the air mixing process between the upper dome space of the containment atmosphere during LOCA operations. They also prevent any hot spot air pockets during the containment cooling mode and avoid any hydrogen concentration in pocket areas. Two dome air circulator trains are required to be operable. Each train consists of two fans with their own motors and controls and is automatically initiated by a containment cooling actuation signal (CCAS). While each train has two fans, only one operable fan is required for the train to be operable, since each fan can provide the necessary flow rate to adequately mix the containment atmosphere. TS 3.6.8 is applicable in MODES 1 and 2.

The NRC staff has reviewed the licensee's proposed changes for Section 3.6 of the SONGS Units 2 and 3 TSs and has determined that the TSs are only needed to provide the LCOs and SRs necessary to maintain functionality of the containment. These TSs include multiple LCOs addressing containment integrity, which includes: containment pressure, containment air temperature, and containment air locks, which forms part of the containment pressure boundary; and containment isolation valves, designed to isolate the containment in the event of a LOCA to prevent the release of fission products to the atmosphere; containment spray and cooling, which limit post-accident pressure and temperature in containment; and dome air circulators that help reduce the potential hydrogen concentration pockets in containment following a design basis accident. All of these TSs are related to assuring the integrity of the containment as a fission product boundary. The containment TSs are only important for a reactor authorized to operate or retain irradiated fuel in the reactor vessel. However, because 10 CFR 50.82(a)(2) prohibits the licensee from operating the reactors or placing fuel in the reactor vessels at SONGS Units 2 and 3, the containment SSCs are no longer functional or used in any capacity and the associated TSs are no longer meaningful.

The NRC staff also reviewed the containment TSs proposed for deletion (TS 3.6.1, TS 3.6.2, TS 3.6.3, TS 3.6.4, TS 3.6.5, TS 3.6.6.1, TS 3.6.6.2, and TS 3.6.8), to ensure that these LCOs no longer satisfy the 10 CFR 50.36 criteria for inclusion in TSs, as described in Section 2.1 of this evaluation. The staff notes that these TSs indicate MODES for which the TSs are applicable. MODES, as defined in TSs, correspond to any one inclusive combination of core reactivity condition, power level, average reactor coolant temperature, and reactor vessel head closure bolt tensioning with fuel in the reactor vessel. The reference to MODES for a permanently shutdown and defueled reactor, such as SONGS Units 2 and 3, has no meaning and is not relevant. Because SCE has submitted certifications pursuant to 10 CFR 50.82(a)(2), it is prohibited from operating the reactors or placing fuel in the reactor vessels, and SONGS Units 2 and 3 are no longer in a configuration or a condition under which the TS MODES apply.

Therefore, the NRC staff finds that the licensee's proposed change to delete TS Section 3.6, Containment Systems, is acceptable.

3.7.13 Section 3.7, Plant Systems

Section 3.7 of the SONGS Units 2 and 3 TSs, "Plant Systems," contains the LCOs, Actions, and SRs that provide for appropriate functional capability of balance-of-plant equipment required for safe operation of the facility. This section contains operability requirements related to the steam generators, feedwater system, cooling water, ventilation, and spent fuel storage.

The licensee proposed deletion of the following LCOs in Section 3.7 of the SONGS Units 2 and 3 TSs:

TS 3.7.1, "Main Steam Safety Valves (MSSVs)," specifies the requirements for the MSSVs to ensure they are capable of providing overpressure protection for the secondary system. The MSSVs also provide protection against overpressurizing the RCPB by providing a heat sink for the removal of energy from the RCS, if the preferred heat sink provided by the condenser and circulating water system, is not available. TS 3.7.1 is applicable in MODES 1, 2, and 3.

TS 3.7.2, "Main Steam Isolation Valves (MSIVs)," specifies the requirements for the MSIVs to ensure that they are capable of isolating steam flow from the secondary side of the SGs following a high energy line break (HELB). MSIV closure terminates flow from the unaffected (intact) steam generator. One MSIV is located in each main steam line outside, but close to, containment. The MSIVs are downstream from the MSSVs, atmospheric dump valves, and auxiliary feedwater pump turbine steam supplies to prevent them from being isolated from the SGs by MSIV closure. Closing the MSIVs isolates each SG from the other, and isolates the turbine, steam bypass system, and other auxiliary steam supplies from the steam generators. TS 3.7.2 is applicable in MODES 1, 2, and 3 except when all MSIVs are closed and deactivated.

TS 3.7.3, "Main Feedwater Isolation Valves (MFIVs)," specifies the requirements for MFIVs. The MFIVs isolate main feedwater (MFW) flow to the secondary side of the steam generators following a HELB. Closure of the MFIVs terminates flow to the steam generators, terminating the event for feedwater line breaks (FWLBs) occurring upstream of the MFIVs. The consequences of events occurring in the main steam lines or in the MFW lines downstream of the MFIVs will be mitigated by their closure. Closure of the MFIVs effectively terminates the addition of feedwater to an affected steam generator, limiting the mass and energy release for steam line breaks (SLBs) or FWLBs inside containment, and reducing the cooldown effects for SLBs. TS 3.7.3 is applicable in MODES 1, 2, and 3 except when MFIV is closed and deactivated.

TS 3.7.4, "Atmospheric Dump Valves (ADVs)," specifies the requirements for providing a method for cooling the unit to shutdown cooling system entry conditions, should the preferred heat sink via the steam bypass system to the condenser not be available. This is done in conjunction with the auxiliary feedwater (AFW) system providing cooling water from the condensate storage tank (CST). TS 3.7.4 is applicable in MODES 1, 2, and 3, and in MODE 4 when steam generator is relied upon for heat removal.

TS 3.7.5, "Auxiliary Feedwater (AFW) System," specifies the requirements to ensure that the AFW system automatically supplies feedwater to the steam generators to remove decay heat from the RCS upon the loss of normal feedwater supply. TS 3.7.5 is applicable in MODES 1, 2, and 3, and in MODE 4 when steam generator is relied upon for heat removal.

TS 3.7.6, "Condensate Storage Tank (CST T-121 and T-120)," specifies the requirements to ensure a safety grade source of water to the SGs for removing decay and sensible heat from the RCS. The CSTs provide a passive flow of water, by gravity, to the AFW System. TS 3.7.6 is applicable in MODES 1, 2, and 3, and in MODE 4 when steam generator is relied upon for heat removal.

TS 3.7.7, "Component Cooling Water (CCW) System," specifies the requirements to ensure that the CCW system provides a heat sink for the removal of process and operating heat from safety-related components during a DBA or transient. During normal operation, the CCW system also provides this function for various nonessential components. The CCW System serves as a barrier to the release of radioactive byproducts between potentially radioactive systems and the salt water cooling system, and thus to the environment. TS 3.7.7 is applicable in MODES 1, 2, 3, and 4.

TS 3.7.7.1, "Component Cooling Water (CCW) Safety Related Makeup System," specifies the requirements to ensure a safety-related CCW makeup system is available to maintain the water inventory in the CCW trains during a 7-day post-accident period. The safety-related makeup system is designed to supply water to the CCW trains following loss of normal CCW makeup from the nuclear service water system. For this purpose, sufficient water inventory is contained in the single primary plant makeup (PPMU) storage tank for both CCW trains. From the PPMU tank, water is transferred to the CCW return heads by two safety-related pumps. TS 3.7.7.1 is applicable in MODES 1, 2, 3, and 4.

TS 3.7.8, "Salt Water Cooling (SWC) System," specifies the requirements to ensure that the SWC system provides a heat sink for the removal of process and operating heat from safety-related components during a DBA or transient. During normal operation, and a normal shutdown, the SWC system also provides this function for various safety-related and nonsafety-related components. The safety-related function is covered by TS 3.7.8. TS 3.7.8 is applicable in MODES 1, 2, 3, and 4.

TS 3.7.10, "Emergency Chilled Water (ECW)," specifies the requirements to ensure that the ECW system provides a heat sink for the removal of process and operating heat from selected safety-related air handling systems during a DBA or transient. The design basis of the ECW system is to remove the post-accident heat load from ESF spaces following a DBA coincident with a loss of offsite power. Each train provides chilled water to the HVAC units at the design temperature and flow rate. TS 3.7.10 is applicable in MODES 1, 2, 3, and 4.

TS 3.7.19, "Secondary Specific Activity," specifies the limit on secondary coolant specific activity during power operation to minimize releases to the environment because of normal operation, AOOs, and accidents. The accident analysis of the main steam line break (MSLB) assumes an

initial secondary coolant specific activity used for determining the radiological consequences of the postulated accident. The accident analysis, based on this and other assumptions, shows that the radiological consequences of an MSLB do not exceed the TEDE limit. TS 3.7.19 is applicable in MODES 1, 2, 3, and 4.

The NRC staff has reviewed the proposed changes to TSs 3.7.1 through TS 3.7.10, and TS 3.7.19 and has determined that these TSs are only necessary to assure the operability of certain plant systems during reactor operation. These TSs involve: MSSVs, which provide overpressure protection for the secondary system; MSIVs, which isolate steam flow from the secondary side of the steam generator following a MSLB; MFIVs, which isolate main feedwater flow to the secondary side of the steam generators following a HELB; ADVs, which provide a method for cooling the unit should the condenser not be available; AFW system, which supplies feedwater to the steam generators upon the loss of the normal feedwater supply; CSTs, which provide the preferred source of water to the steam generators for removing decay and sensible heat from the RCS; CCW system, CCW safety-related makeup system, and the SWC system, which provide a heat sink for the removal of process and operating heat from safety-related components during a DBA or transient to the ultimate heat sink; the ECW system that removes heat from ESF spaces through safety-related air handling systems; and secondary specific activity, which specifies a limit on secondary coolant specific activity during power operation.

The above TSs were intended to protect the fuel in the reactor from potential operational transients and accidents. However, 10 CFR 50.82(a)(2) prohibits the licensee from operating the reactor or placing fuel in the reactor vessel. Consequently, there are no longer any transient or accident conditions that these systems and components protect against or mitigate. Therefore, the NRC staff finds the deletion of 3.7.1 through 3.7.10, as detailed above, is acceptable. TS 3.7.19 provides the operational limits on secondary coolant specific activity limiting the potential radiological consequences of an accident that could release pressurized steam from the SGs. Since 10 CFR 50.82(a)(2) prohibits the licensee from operating the reactor or placing fuel in the reactor vessel, there is no source of heat available to pressurize the SGs and no source of activity. Therefore, the NRC staff finds the deletion of TS 3.7.19 is acceptable.

The NRC staff has also reviewed Section 3.7 of the SONGS Units 2 and 3, TS 3.7.1, TS 3.7.2, TS 3.7.3, TS 3.7.4, TS 3.7.5, TS 3.7.6, TS 3.7.7, TS 3.7.7.1, TS 3.7.8, TS 3.7.10, and TS 3.7.19 proposed for deletion to ensure that these LCOs no longer satisfy the 10 CFR 50.36 criteria for inclusion in TSs, as described in Section 2.1 of this evaluation. The staff notes that these TSs indicate MODES for which the TSs are applicable. MODES, as defined in TSs, correspond to any one inclusive combination of core reactivity condition, power level, average reactor coolant temperature, and reactor vessel head closure bolt tensioning with fuel in the reactor vessel. The reference to MODES for permanently shutdown and defueled reactors, such as SONGS Units 2 and 3, has no meaning and is not relevant. Because SCE has submitted certifications pursuant to 10 CFR 50.82(a)(2), it is prohibited from operating the reactors or placing fuel in the reactor vessels and SONGS Units 2 and 3 are no longer in a configuration or a condition under which the TS MODES apply.

Therefore, based on the evaluation above, the NRC staff finds that the licensee's proposed change to delete Plant Systems TS 3.7.1, TS 3.7.2, TS 3.7.3, TS 3.7.4, TS 3.7.5, TS 3.7.6, TS 3.7.7, TS 3.7.7.1, TS 3.7.8, TS 3.7.10 and 3.7.19, is acceptable.

TS 3.7.11, "Control Room Emergency Air Cleanup System (CREACUS)," specifies the requirements to ensure that the CREACUS provides a protected environment from which operators can control SONGS Units 2 and 3, following an uncontrolled release of radioactivity, hazardous chemicals, or smoke. The CREACUS consists of two independent, redundant trains that recirculate and filter the air in the control room envelope (CRE) and a CRE boundary that limits the inleakage of unfiltered air. Each CREACUS train consists of an emergency air conditioning unit, emergency ventilation air supply unit, emergency isolation dampers, and cooling coils and two cabinet coolers. Each emergency air conditioning unit includes a prefilter, a high efficiency particulate air (HEPA) filter, an activated carbon adsorber section for removal of gaseous activity (principally iodines), and a fan. A second bank of HEPA filters follows the adsorber section to collect carbon fines. Ductwork, motor-operated dampers, doors, barriers, and instrumentation also form part of the system. Upon receipt of the actuating signal, normal air supply to the CRE is isolated and the stream of ventilation air is recirculated through the system's filter trains. The prefilters remove any large particles in the air to prevent excessive loading of the HEPA filters and charcoal adsorbers.

There are two CREACUS operational modes. Emergency mode is an operational mode when the control room is isolated to protect operational personnel from radioactive exposure through the duration of a DBA. Isolation mode is an operational mode when the CRE is isolated to protect operational personnel from toxic gases and smoke. Actuation of the CREACUS places the system into either of two separate states of operation, depending on the initiation signal. Actuation of the system to either the emergency mode or isolation mode of CREACUS operation closes the unfiltered-outside-air intake and unfiltered exhaust dampers and aligns the system for recirculation of air within the CRE through the redundant trains of HEPA and charcoal filters. The emergency mode also initiates pressurization of the CRE. Outside air is added to the air being recirculated from the CRE. Pressurization of the CRE minimizes infiltration of unfiltered air through the CRE boundary from all the surrounding areas adjacent to the CRE boundary. The CRE supply and the outside air supply of the normal control room HVAC are monitored by radiation and toxic-gas detectors, respectively. One detector output above the setpoint will cause actuation of the emergency mode or isolation mode as required. The actions of the isolation mode are more restrictive, and will override the actions of the emergency mode of operation. TS 3.7.11 is applicable in MODES 1, 2, 3, 4, 5, and 6 and during movement of fuel assemblies in the containment or fuel storage pool.

When SONGS Units 2 and 3 were authorized to operate, the CREACUS provided a protected environment from which operators could control the units following postulated accidents involving an uncontrolled release of radioactivity, including an FHA. Prior to SONGS Units 2 and 3 permanently shutting down and defueling, the TSs for the CREACUS provided the LCOs and SRs necessary to maintain the control room environment following an accident. Specifically, during irradiated fuel movement, the CREACUS provided a protected environment from which operators can control the unit following a postulated uncontrolled release of radioactivity from an FHA. CRE will remain habitable during and following a DBA. In MODES 5

and 6, the CREACUS is required to cope with the release from a rupture of a waste gas tank. During movement of fuel assemblies, the CREACUS must be operable to cope with the release from an FHA.

The licensee provided information on the toxic gases isolation of CREACUS in Section 3.2.10.2.3 of the licensee's permanently defueled technical specification amendment request. Specifically, per the NRC's SE associated with the issuance of SONGS License Amendment Nos. 127 and 116 for Units 2 and 3, respectively dated February 9, 1996 (ADAMS Accession No. ML021990684), the toxic gas isolation of CREACUS is not relied on to prevent or mitigate a design basis accident or transient because the plant design includes other means to safely shut down the plant if the control room becomes uninhabitable. As such, the toxic gas isolation instrumentation was relocated from the TS and placed in the Licensee Controlled Specifications with an applicability of MODES 1, 2, 3, 4, 5, and 6. Since an NRC SE has already accepted the removal of toxic gas isolation of CREACUS from the TSs, a new NRC staff determination is not required. The staff concludes that automatic toxic gas isolation of CREACUS is not required during movement of fuel assemblies in the fuel storage pool at the permanently defueled SONGS Units 2 and 3.

With the termination of reactor operations at SONGS Units 2 and 3 and the permanent removal of the fuel from the reactor core in each unit, the postulated accidents involving failure or malfunction of the reactor, RCS, or secondary system are no longer applicable. While there are no transients that continue to apply to SONGS Units 2 and 3, there are still postulated DBAs. As discussed in Sections 3.1 through 3.6 of this SE, the remaining DBAs applicable to the defueled reactors of SONGS Units 2 and 3 show that the dose consequences are acceptable without relying on SSCs remaining functional for accident mitigation during and following the event, with the exception of the SFP structure.

The NRC staff evaluated these accident analyses and confirmed that no ESF system is used to mitigate the CR, EAB, or LPZ dose consequences. This includes no credit for the FHIS, the PACU filtration system, the CRIS and the CREACUS. Since SONGS Units 2 and 3 are permanently shutdown and defueled, and greater than 17 months of decay time has elapsed since permanent shut down, the remaining DBAs applicable to the facility demonstrate that the dose consequences within the CR are acceptable without relying on SSCs remaining functional for accident mitigation, including an FHA in the FHB. (The one exception to this is the continued function of the passive fuel storage pool structure, which will be maintained as a TS for SONGS Units 2 and 3.)

In summary, the radiological consequences of the remaining DBAs for SONGS Units 2 and 3 assume no credit for CR isolation or recirculation filtration and no credit for any accident mitigation by the FHB ventilation system. Calculated doses at the EAB, LPZ, and CR are within 10 CFR Part 50.67 limits, and RG 1.183 dose limits. Since the DBA accident analysis for SONGS Units 2 and 3 assumed no credit for control room post-accident recirculation system emergency ventilation or filtration, the CREACUS is no longer required. Therefore, isolation of the CRE via the CRIS and CREACUS is not necessary for any of the postulated DBAs. As noted before, the intent of Criterion 3 of 10 CFR 50.36(c)(2)(ii)(C) is to capture into TSs those SSCs that are part of the primary success path of a safety sequence analysis. With SONGS

Units 2 and 3 permanently shutdown and defueled, and the irradiated fuel having decayed for a minimum period of 17 months, the CREACUS is no longer needed or credited in the primary success path of a safety sequence analysis related to an accident. Since the radiological consequences of the accident analyses are within the appropriate acceptance criteria without credit for the CREACUS, the NRC staff finds that the licensee's proposed change to delete TS 3.7.11, is acceptable.

The licensee intends to retain TS 3.7.16, TS 3.7.17, and TS 3.7.18 but revise these TSs to delete the REQUIRED ACTIONS note that states that LCO 3.0.3 is not applicable. LCO 3.0.3 is being deleted from the SONGS Units 2 and 3 TSs and removal of a reference to TS 3.0.3 is a conforming change. In addition, with the deletion of all TSs in Section 3.1 through 3.6, the licensee also proposes to renumber these TSs to 3.1.1 to 3.1.3, respectively

TS 3.7.16, "Fuel Storage Pool Water Level," specifies the requirements to ensure that the minimum water level in the SFP meets the assumptions of iodine decontamination factors following an FHA. The water also provides shielding during the movement of spent fuel. This TS is applicable during movement of irradiated fuel assemblies in the SFP. The licensee has proposed to retain this TS in the permanently defueled TSs essentially unchanged. The Note in Required Action A.1 (LCO 3.0.3 is not applicable), is being deleted to conform to the deletion of TS LCO 3.0.3. The licensee has also proposed to renumber this TS as 3.1.1, based on the proposed deletion of all the preceding TSs.

Criterion 2 of 10 CFR 50.36(c)(2)(ii)(B) states that TS LCOs must be established for a "process variable, design feature, or operating restriction that is an initial condition of a design basis accident or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier." The purpose of this criterion is to capture those process variables that have initial values assumed in the DBA. TS 3.7.16, "Fuel Storage Pool Water Level," specifies the TS required LCOs and SRs that ensure the minimum water level in the SFP meets the assumptions of iodine decontamination factors following an FHA or cask drop accident.

SCE's analysis of the postulated FHA or cask drop accident assumes that there is at least 23 feet of water between the top of the damaged fuel assemblies and the fuel pool surface. The gap activity in the damaged rods is assumed to be instantaneously released into the SFP. Radionuclides in the gap release are assumed to be filtered by the 23 feet of water before emerging from the SFP. The activity exhaust rate from the auxiliary building is established to complete the release in 2 hours, as required by RG 1.183, but does not credit the auxiliary building ventilation for any mitigation of the release.

Since the 23-foot water level of the SFP is an initial condition of the FHA and the cask drop DBA, it satisfies Criterion 2 for inclusion in TSs and is being retained for SONGS Units 2 and 3 in their permanently shutdown and defueled condition. The amendment request by SCE does not involve any change to the technical language in the TS. The discussion in this evaluation of the SFP water level TS is provided only for completeness since the SFP water level is an important initial condition in the FHA and cask drop accident analysis and will continue to be part of the SONGS TSs.

TS 3.7.17, "Fuel Storage Pool Boron Concentration," specifies the requirements to ensure that the SFP boron concentration is > 2000 parts per million (ppm). The specified concentration of dissolved boron in the SFP preserves the assumptions used in the analyses of the potential critical accident scenarios as described in the criticality analysis of record, which is that a minimum of 2000 ppm of boron is needed to ensure that criticality does not occur during the worst case fuel loading accident. This concentration of dissolved boron is the minimum required for fuel assembly storage and movement within the spent fuel pool. This TS is applicable whenever fuel assemblies are stored in the spent fuel pool. This TS is being retained in the permanently defueled TS essentially unchanged. The Note in Required Action A.1 (LCO 3.0.3 is not applicable), is being deleted to conform to the deletion of TS LCO 3.0.3. The licensee has also proposed to renumber this TS as 3.1.2, based on the proposed deletion of all the preceding TSs.

TS 3.7.18, "Spent Fuel Assembly Storage," specifies the restrictions on the placement of fuel assemblies within the SFP, in accordance with Figure 3.1.3-1 through Figure 3.1.3-4 in the accompanying LCO, to ensure the keff of the SFP will always remain < 0.95, assuming the pool to be flooded with unborated water. This TS applies whenever any fuel assembly is stored in the spent fuel pool. TS 3.7.18 is being retained in the permanently defueled TS essentially unchanged. The Note in Required Action A.1 (LCO 3.0.3 is not applicable), is being deleted to conform to the deletion of TS LCO 3.0.3. The licensee has also proposed to renumber this TS as 3.1.3, based on the proposed deletion of all the preceding TSs.

The NRC staff reviewed the proposed deletion of the reference to LCO 3.0.3 in the Required Actions Note in TS 3.7.16, TS 3.7.17, and TS 3.7.18. The staff finds that deletion of the Note, "LCO 3.0.3 is not applicable," in each of these TSs is appropriate and the conforming change to the deletion of TS LCO 3.0.3, as discussed in Section 3.7.6 of this SE. Therefore, the staff finds that the licensee's proposed change to delete the reference to LCO 3.0.3 in TS 3.7.16, TS 3.7.17, and TS 3.7.18 (renumber as TS 3.1.1, TS 3.1.2, and TS 3.1.3, respectively – see below), is acceptable.

The NRC staff also reviewed the proposed change to renumber TS 3.7.16, 3.7.17, and TS 3.7.18, to TS 3.1.1, TS 3.1.2, and TS 3.1.3, respectively, and found the change to be editorial and conforming to the overall changes to the TSs. Therefore, the NRC staff finds that the licensee's proposed renumbering of the TSs is acceptable.

3.7.14 Section 3.8, Electrical Power Systems

The licensee proposed to delete SONGS Units 2 and 3 electrical power systems TS 3.8.1, TS 3.8.4, TS 3.8.7, and TS 3.8.9, since these TSs are MODE dependent and only applicable to an operating reactor. Therefore, these TSs do not apply to the permanently shutdown and defueled condition of SONGS Units 2 and 3.

TS 3.8.1, "AC [Alternating Current] Sources - Operating," specifies the requirements to ensure that the offsite power sources (normal preferred and alternate preferred power sources), and the standby power sources (Train A and Train B DGs), provide sufficient capacity, capability, redundancy, and reliability to ensure the availability of necessary power to ESF systems so that

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the fuel, RCS, and containment design limits are not exceeded. TS 3.8.1 is applicable in MODES 1, 2, 3, and 4.

TS 3.8.4, "DC [Direct Current] Sources - Operating," specifies the requirements to ensure that the DC electrical power subsystems (with each subsystem consisting of one battery, the required battery charger, and the corresponding control equipment and interconnecting cabling supplying power to the associated bus within the subsystem) are required to be OPERABLE to ensure the availability of the required power to shut down the reactor and maintain it in a safe condition after an AOO or postulated DBA. TS 3.8.4 is applicable in MODES 1, 2, 3, and 4.

TS 3.8.7, "Inverters - Operating," specifies the requirements to ensure that required inverters are OPERABLE such that the redundancy incorporated into the design of the RPS and ESFAS instrumentation and controls is maintained. These requirements include the required capacity, capability, redundancy, and reliability to ensure the availability of necessary power to the RPS and ESFAS instrumentation and controls so that the fuel, RCS, and containment design limits are not exceeded. TS 3.8.7 is applicable in MODES 1, 2, 3, and 4.

TS 3.8.9, "Distribution Systems - Operating," specifies the requirements to ensure availability of AC, DC, and AC instrument bus electrical power for the systems required to shut down the reactor and maintain it in a safe condition after an AOO or a postulated DBA. The AC, DC, and AC vital electrical power distribution systems are designed to provide sufficient capacity, capability, redundancy, and reliability to ensure the availability of necessary power to ESF systems so that the fuel, RCS, and containment design limits are not exceeded. TS 3.8.9 is applicable in MODES 1, 2, 3, and 4.

The NRC staff has reviewed the SONGS Units 2 and 3 electrical power systems LCOs in TS 3.8.1, TS 3.8.4, TS 3.8.7, and TS 3.8.9, which have been proposed for deletion, to ensure that these LCOs no longer satisfy the 10 CFR 50.36 criteria for inclusion in TSs, as described in Section 2.1 of this evaluation. The staff notes that these TSs indicate MODES for which the TS is applicable. MODES, as defined in TSs, correspond to any one inclusive combination of core reactivity condition, power level, average reactor coolant temperature, and reactor vessel head closure bolt tensioning with fuel in the reactor vessel. The reference to MODES for the permanently shutdown and defueled reactors, such as SONGS Units 2 and 3, has no meaning and is not relevant. Because SCE has submitted certifications pursuant to 10 CFR 50.82(a)(2), it is prohibited from operating the reactors or placing fuel in the reactor vessels. Therefore, SONGS Units 2 and 3 are no longer in a configuration or a condition under which these TS MODES apply. Based on the above, the staff finds the deletion of TS 3.8.1, TS 3.8.4, TS 3.8.7, and TS 3.8.9 from TS Section 3.8, Electrical Systems, is acceptable.

The licensee has also proposed to delete SONGS Units 2 and 3 electrical power systems TS 3.8.2, TS 3.8.3, TS 3.8.5, TS 3.8.6, TS 3.8.8, and TS 3.8.10, based on the MODE dependent applicably of these TSs. However, these TSs are also directly applicable during the movement of irradiated fuel assemblies or are support systems for TSs required during the movement of irradiated fuel assemblies. The following evaluations of these electrical power systems TSs assess the licensee's justification as to why these TSs do not apply to the

permanently shutdown and defueled condition of SONGS Units 2 and 3 during movement of irradiated fuel assemblies.

TS 3.8.2, "AC Sources - Shutdown," specifies the requirements to ensure that the offsite power sources (normal preferred and alternate preferred power sources), and the standby power sources (Train A and Train B DGs), provide sufficient capacity, capability, redundancy, and reliability to ensure the availability of necessary power to ESF systems so that the fuel, RCS, and containment design limits are not exceeded. TS 3.8.2 is applicable during MODES 5 and 6 and during movement of fuel assemblies in containment or in the fuel storage pool.

TS 3.8.3, "Diesel Fuel Oil, Lube Oil, and Starting Air," provides for proper operation of the DGs, by specifying the parameters and ensuring there will be sufficient quantity and proper quality of the fuel oil, lube oil, and starting air systems. Stored diesel fuel oil is required to have sufficient supply for 7 days of rated load operation for each DG. It is also required to meet specific standards for quality. Additionally, sufficient lubricating oil supply must be available to ensure the capability to operate each DG at rated load for 7 days. Lastly, each DG is equipped with two air start systems, which have adequate capacity for five successive start attempts on the DG without recharging the air start receivers. TS 3.8.3 is applicable whenever the DGs are required to be operable.

TS 3.8.5, "DC Sources Shutdown," specifies the requirements to ensure availability of the DC electrical power system and subsystems (with each subsystem consisting of one battery, required battery charger, and the corresponding control equipment and interconnecting cabling supplying power to the associated bus within the subsystem), in order to provide normal and emergency DC electrical power for the DGs, emergency auxiliaries, and control and switching. TS 3.8.5 is applicable during MODES 5 and 6 and during movement of fuel assemblies in containment or in the fuel storage pool.

TS 3.8.6, "Battery Parameters," specifies the requirements to ensure the limits on battery float current as well as electrolyte temperature, level, and float voltage for the DC power subsystem batteries. Battery parameters are required solely for the support of the associated DC electrical power subsystems (per TS 3.8.4 and TS 3.8.5). Therefore, battery parameter limits are only required (and TS 3.8.6 is only applicable) when the DC electrical power source is required to be operable.

TS 3.8.8, "Inverters – Shutdown," specifies the requirements to ensure stability and reliability of the preferred source of power for the 120 Volt AC vital buses. The inverters can be powered from an internal AC source/rectifier or from the station battery. The inverter provides an uninterruptible power source for the safety-related instrumentation and controls. TS 3.8.8 is applicable during MODES 5 and 6 and during movement of fuel assemblies within containment or in the fuel storage pool.

TS 3.8.10, "Distribution Systems - Shutdown," specifies the requirements for the onsite AC, DC, and AC instrument bus electrical power distribution systems. The TS specifies sufficient capacity, capability, redundancy, and reliability of the distribution system to ensure the availability of necessary power to ESF systems so that the fuel, RCS, and containment design

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limits are not exceeded. TS 3.8.10 is applicable during MODES 5 and 6 and during movement of fuel assemblies within containment or in the fuel storage pool.

The SONGS Unit 2 and 3 TS Basis documents indicates that the shutdown electrical power systems TS 3.8.2, TS 3.8.3, TS 3.8.5, TS 3.8.6, TS 3.8.8, and TS 3.8.10 provide assurance that:

a. The units can be maintained in the shutdown or refueling condition for extended periods;

b. Sufficient instrumentation and control capability is available for monitoring and maintaining the units status; and

c Adequate AC electrical power is provided to mitigate events postulated during shut down, such as an FHA.

The NRC staff has reviewed the SONGS Units 2 and 3 shutdown electrical power systems TSs (TS 3.8.2, TS 3.8.3, TS 3.8.5, TS 3.8.6, TS 3.8.8, and TS 3.8.10), which have been proposed for deletion, to ensure that these LCOs no longer satisfy the 10 CFR 50.36 criteria for inclusion in TSs, as described in Section 2.1 of this evaluation. The staff notes that these TSs indicate MODES for which the TS is applicable. MODES, as defined in TSs, correspond to any one inclusive combination of core reactivity condition, power level, average reactor coolant temperature, and reactor vessel head closure bolt tensioning with fuel in the reactor vessel. The reference to MODES for the permanently shutdown and defueled reactors, such as SONGS Units 2 and 3, has no meaning and is not relevant. Because SCE has submitted certifications pursuant to 10 CFR 50.82(a)(2), it is prohibited from operating the reactors or placing fuel in the reactor vessels and SONGS Units 2 and 3 is no longer in a configuration or a condition under which these TS MODES apply. Based on the above, the NRC staff finds that the licensee's proposed change to delete TS 3.8.2, TS 3.8.3, TS 3.8.5, TS 3.8.6, TS 3.8.8, and TS 3.8.10, from TS Section 3.8, Electrical Systems, for MODES 5 and 6, is acceptable.

The NRC staff also reviewed the non-MODE dependent applicability during movement of fuel assemblies within containment and in the fuel storage pool. Because 10 CFR 50.82(a)(2) prohibits operation of the plant or placing fuel in the reactor vessel, the prohibition on placing fuel in the reactor vessel, it also precludes core alterations and the movement of fuel assemblies within containment. Therefore, these TSs are no longer needed during movement of fuel assemblies within containment.

As detailed in Sections 3.1 through 3.6 of this SE, the remaining accident analyses applicable to the permanently shutdown and defueled reactors of SONGS Units 2 and 3 show that the dose consequences within the control room are acceptable without relying on SSCs remaining functional for accident mitigation during any of the remaining DBAs, including FHAs. (The one exception to this is the continued function of the passive fuel storage pool structure, which will be maintained as a TS for SONGS).

For TS 3.8.2, AC Sources – Shutdown, the FHA is the applicable DBA related to the TS requirement for functional capability of AC sources (offsite power and DGs) during the TS

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specified condition of during movement of fuel assemblies in the fuel storage pool. Because the FHA analysis, and the other DBAs identified for SONGS Units 2 and 3, do not rely on normal or emergency power for accident mitigation (including any need for providing airborne radiological protection), the AC sources are not required during movement of fuel assemblies in the fuel storage pool for mitigation of a potential FHA or any of the other DBAs. Specifically, the accident analyses show that the dose consequences are acceptable without relying on any SSCs to remain functional during and following the postulated events, with the exception of the SFP support structure. Therefore, during movement of fuel assemblies in the fuel storage pool, there are no systems that function or actuate and are credited as part of the initial conditions of an analysis or as part of the primary success path for mitigation of the DBA. As such, the requirement for AC sources is no longer necessary because there are no design-basis events that rely on AC sources for mitigation. Consequently, AC sources no longer meet the requirements of Criterion 3 of 10 CFR 50.36(c)(2)(ii)(C) and can be removed from TSs. Therefore, the NRC staff finds that the licensee's proposed change to delete TS 3.8.2, during movement of fuel assemblies in the fuel storage pool, is acceptable.

The NRC staff has reviewed the need for TS 3.8.3, Diesel Fuel Oil, Lube Oil, and Starting Air, during movement of fuel assemblies in the fuel storage pool. Since TS 3.8.3 exists solely to support the DG requirements of TS 3.8.1 and TS 3.8.2, the deletion of these TSs is consistent with the elimination of the need for DGs and also eliminates the need for the DG support systems. The NRC staff has determined that the requirement for DGs and associated supporting TSs are no longer necessary because the remaining DBAs for SONGS Units 2 and 3 do not rely on the DGs for mitigation. Therefore, the NRC staff finds that the licensee's proposed change to delete TS 3.8.3, during movement of fuel assemblies in the fuel storage pool, is acceptable.

The NRC staff has reviewed the need for TS 3.8.5, DC Sources – Shutdown, during movement of fuel assemblies in the fuel storage pool. Because the FHA analysis, and the other DBAs identified for SONGS Units 2 and 3, do not rely on safety-related DC sources of electrical power for accident mitigation (including any need for providing airborne radiological protection), the DC sources are not required during movement of fuel assemblies in the fuel storage pool for mitigation of a potential FHA or any of the other DBAs. Specifically, the accident analyses show that the dose consequences are acceptable without relying on any SSCs to remain functional during and following the postulated events. Therefore, during movement of fuel assemblies in the fuel storage pool, there are no systems that function or actuate and are credited as part of the initial conditions of an analysis or as part of the primary success path for mitigation of the DBA. As such, the requirement for DC sources is no longer necessary because there are no design-basis events that rely on DC sources for mitigation. Consequently, DC sources no longer meet the requirements of Criterion 3 of 10 CFR 50.36(c)(2)(ii)(C) and can be removed from TSs. Therefore, the NRC staff finds that the licensee's proposed change to delete TS 3.8.5, during movement of fuel assemblies in the fuel storage pool, is acceptable.

The NRC staff has reviewed the need for TS 3.8.6, Battery Parameters, during movement of fuel assemblies in the fuel storage pool. Since TS 3.8.6 exists solely to support the DC source requirements of TS 3.8.4 and TS 3.8.5, the deletion of these TSs is consistent with the elimination of the need for DC sources and also obviates the need for the battery support

systems. The staff has determined that the requirement for DC sources and associated supporting TSs are no longer necessary because the remaining DBAs for SONGS Units 2 and 3 do not rely on the DC sources for mitigation. Therefore, the NRC staff finds that the licensee's proposed change to delete TS 3.8.6, battery parameters, during movement of fuel assemblies in the fuel storage pool, is acceptable.

The NRC staff has reviewed the need for TS 3.8.8, Inverters – Shutdown, during movement of fuel assemblies in the fuel storage pool. Because the FHA analysis, and the other DBAs identified for SONGS Units 2 and 3 do not rely on inverters or the safety-related 120 Volt AC electrical power for accident mitigation (including any need for providing airborne radiological protection), the inverters are not required during movement of fuel assemblies in the fuel storage pool for mitigation of a potential FHA or any of the other DBAs. Specifically, the accident analyses show that the dose consequences are acceptable without relying on any SSCs to remain functional during and following the postulated events, with the exception of the SFP support structure. Therefore, during movement of fuel assemblies in the fuel storage pool, there are no systems that function or actuate and are credited as part of the initial conditions of an analysis or as part of the primary success path for mitigation of the DBA. As such, the requirement for inverters is no longer necessary because there are no design-basis events that rely on inverters for mitigation. Consequently, inverters no longer meet the requirements of Criterion 3 of 10 CFR 50.36(c)(2)(ii)(C) and can be removed from TSs. Therefore, the NRC staff finds that the licensee's proposed change to delete TS 3.8.8, during movement of fuel assemblies in the fuel storage pool, is acceptable.

The NRC staff has reviewed the need for TS 3.8.10, Distribution System – Shutdown, during movement of fuel assemblies in the fuel storage pool. Because the FHA analysis, and the other DBAs identified for SONGS Units 2 and 3, do not rely on the safety-related AC, DC and AC instrument bus electrical distribution systems for accident mitigation (including any need for providing airborne radiological protection), these safety-related distributions systems are not required during movement of fuel assemblies in the fuel storage pool for mitigation of a potential FHA or any of the other DBAs. Specifically, the accident analyses show that the dose consequences are acceptable without relying on any SSCs to remain functional during and following the postulated events, with the exception of the SFP support structure. Therefore, during movement of fuel assemblies in the fuel storage pool, there are no systems that function or actuate and are credited as part of the initial conditions of an analysis or as part of the primary success path for mitigation of the DBA. As such, the requirement for safety-related AC, DC and AC instrument bus electrical distribution systems is no longer necessary because there are no design-basis events that rely on safety-related AC, DC and AC instrument bus electrical distribution systems for mitigation. Consequently, AC, DC and AC instrument bus electrical distribution systems no longer meet the requirements of Criterion 3 of 10 CFR 50.36(c)(2)(ii)(C) and can be removed from TSs. Therefore, the NRC staff finds that the licensee's proposed change to delete TS 3.8.10, during movement of fuel assemblies in the fuel storage pool, is acceptable.

3.7.15 Section 3.9, Refueling Operations

Section 3.9 of the SONGS Units 2 and 3 TSs, "Refueling Operations," contains the LCOs, Actions, and SRs related to refueling operations. This section contains the following LCOs:

TS 3.9.1, "Boron Concentration," places limits on the boron concentrations of the RCS and the refueling canal to ensure that the reactor remains subcritical during refueling. Refueling boron concentration is the soluble boron concentration in the coolant in each of these volumes, which have direct access to the reactor core during refueling. The boron concentration limits required by TS LCO 3.9.1 are specified in the COLR. The boron concentration limit specified in the COLR will maintain a k_{eff} of < 0.95 during fuel handling operations with CEAs and fuel assemblies assumed to be in the most adverse configuration (least negative reactivity) allowed by unit procedures. TS 3.9.1 is applicable in MODE 6.

TS 3.9.2, "Nuclear Instrumentation," requires that two source range monitors (SRMs) to be OPERABLE to ensure that redundant monitoring capability is available to detect changes in core reactivity. The SRMs are required to provide a signal to alert the operator to unexpected changes in core reactivity such as by a boron dilution event or an improperly loaded fuel assembly. TS 3.9.2 is applicable in MODE 6.

TS 3.9.3, "Containment Penetrations," specifies the requirements for containment closure during the conduct of CORE ALTERATIONS and movement of fuel assemblies within containment. The containment penetrations included within TS 3.9.3 are the equipment hatch, personnel airlock doors, and penetrations that provide direct access from the containment atmosphere to the outside atmosphere. TS 3.9.3 limits the consequences of an FHA involving handling fuel within containment by limiting the potential escape paths for fission product radioactivity released within containment. TS 3.9.3 is applicable during CORE ALTERATIONS and during the movement of fuel assemblies within containment.

TS 3.9.4, "Shutdown Cooling (SDC) and Coolant Circulation - High Water Level," specifies requirements for the SDC system in MODE 6 to remove decay heat and sensible heat from the RCS, to provide mixing of borated coolant, to provide sufficient coolant circulation to minimize the effects of a boron dilution accident, and to prevent boron stratification. One loop of the SDC system is required to be OPERABLE and in operation in MODE 6, with the water level > 20 feet above the top of the reactor vessel flange. Only one SDC loop is required to be OPERABLE, because the volume of water above the reactor vessel flange provides backup decay heat removal capability. TS 3.9.4 is applicable in MODE 6, with the water level > 20 feet above the top of the reactor vessel flange.

TS 3.9.5, "Shutdown Cooling (SDC) and Coolant Circulation - Low Water Level," also specifies requirements for the SDC system in MODE 6 to remove decay heat and sensible heat from the RCS, to provide mixing of borated coolant, to provide sufficient coolant circulation to minimize the effects of a boron dilution accident, and to prevent boron stratification. However, with the water level < 20 feet above the top of the reactor vessel flange, both SDC loops must be OPERABLE. Additionally, one loop of SDC must be in operation. TS 3.9.5 is applicable in MODE 6 with the water level < 20 feet above the top of the reactor vessel flange.

TS 3.9.6, "Refueling Water Level," specifies a minimum water level of 23 feet above the top of the reactor vessel flange during movement of fuel assemblies or CEAs within the reactor pressure vessel, and during movement of fuel assemblies within containment. A minimum refueling cavity water level of 23 feet above the top of the reactor vessel flange is required to ensure that the radiological consequences of a postulated FHA inside containment are within acceptable limits. The requirements of TS LCO 3.9.6, in conjunction with a minimum decay time of 72 hours prior to fuel movement, ensures that the release of fission product radioactivity, subsequent to an FHA, results in doses that are well within the guideline values specified in Regulatory Guide 1.183. TS 3.9.6 is only applicable during movement of fuel assemblies or CEAs within the reactor pressure vessel, and during movement of fuel assemblies within containment.

The licensee proposed to delete Section 3.9 of the SONGS Units 2 and 3 TSs LCOs, since they are only applicable to an operating reactor and do not apply to the permanently shutdown and defueled condition of SONGS Units 2 and 3.

The NRC staff has reviewed the proposed changes and has determined that Section 3.9 TSs are only needed to provide the LCOs and SRs necessary to maintain functionality of plant systems required for refueling operations. These TSs involve: boron concentration, which places limits on the boron concentrations of the RCS and the fuel transfer canal during refueling; nuclear instrumentation, which monitors the core reactivity condition during refueling operations; containment penetrations, which specifies requirements for containment closure during the conduct of refueling operations; residual heat removal and coolant circulation – high and low water level, which removes decay heat and sensible heat from the RCS, provides mixing of borated coolant, and prevents boron stratification; and refueling cavity water level, which specifies a minimum water level of 23 feet above the top of the reactor vessel flange during movement of irradiated fuel assemblies within containment. However, 10 CFR 50.82(a)(2) prohibits the licensee from operating the plant or placing fuel in the reactor vessel. Therefore, refueling operations are no longer permitted at SONGS Units 2 and 3, and the LCOs in Section 3.9 TSs are no longer relevant.

The NRC staff has also reviewed the refueling operations TSs proposed for deletion to ensure that these LCOs were no longer required to satisfy the 10 CFR 50.36 criteria for inclusion in TSs, as described in Section 2.1 of this evaluation. The NRC staff notes that these TSs indicate MODES for which each TS is applicable. MODES, as defined in TSs, correspond to any one inclusive combination of core reactivity condition, power level, average reactor coolant temperature, and reactor vessel head closure bolt tensioning with fuel in the reactor vessel. The reference to MODES for a permanently shutdown and defueled reactor has no meaning and is not relevant. Because SCE has submitted certifications pursuant to 10 CFR 50.82(a)(2), it is prohibited from operating the reactor or placing fuel in the reactor vessel and SONGS Units 2 and 3 are no longer in a configuration or a condition under which the TS MODES apply.

Based on the above, the NRC staff finds that the licensee's proposed change to delete TS Section 3.9, Refueling Operations, is acceptable.

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3.7.16 Section 4.0, Design Features

TS 4.1, "Site" provides a description regarding the location of SONGS. The licensee has proposed to retain this TS section in the permanently defueled SONGS Units 2 and 3 TSs with no changes.

TS 4.2, "Reactor Core," provides a general description of the number of and design material requirements for the fuel and control element assemblies used in the reactor core. The licensee has proposed to delete the design feature descriptions for fuel and control element assemblies, since they are only applicable to an operating reactor and do not apply to the permanently shutdown and defueled condition of SONGS Units 2 and 3.

The NRC staff has reviewed the proposed changes to delete the reactor core fuel and control element assemblies design features from SONGS Units 2 and 3 TSs. Since 10 CFR 50.82(a)(2) prohibits the licensee from operating the reactors or placing fuel in the reactor vessels, the design features related to the reactor core fuel assemblies and control rods are no longer relevant at SONGS Units 2 and 3. Therefore, the NRC staff finds that the licensee's proposed change to delete TS 4.2, reactor core design features, is acceptable.

TS 4.3, "Fuel Storage," provides a description and the requirements regarding prevention of criticality of spent fuel, prevention of SFP drainage and spent fuel capacity limitations. This TS section is being retained in the permanently defueled TSs, with the exception of TS 4.3.1.2, which is the design and maintenance of the new fuel storage racks as discussed below. The licensee has also made editorial changes to the TS references in this section to conform to the proposed renumbering of certain retained TSs.

TS 4.3.1.2 has been proposed to be deleted because new fuel is no longer stored onsite and License Condition 2.B.(3) is being revised to no longer allow receipt of new fuel. The NRC staff has reviewed the proposed changes to remove the new fuel storage rack design features from the TSs. Since the licensee currently has no new fuel stored onsite and since the facility license will no longer allow new fuel to be stored onsite, the requirements for new fuel storage racks are no longer applicable.

Based on the above, the NRC staff finds the proposed changes to delete the new fuel storage rack design features from SONGS Units 2 and 3 TS 4.3.1.2 to be acceptable. The staff also reviewed the proposed renumbering of references in TS 4.3.1, Criticality, and determined that the changes to be conforming and editorial in nature. Therefore, the NRC staff finds that the licensee's proposed changes to TS 4.3, Fuel Storage, is acceptable.

3.7.17 Section 5.2, Organization and Section 5.3, Facility Staff Qualifications

SONGS Units 2 and 3 permanently defueled TS 5.1, "Responsibility"; TS 5.2, "Organization"; and TS 5.3, "Facility Staff Qualifications," were previously approved by the NRC staff in License Amendment Nos. 227 and 220 for SONGS Units 2 and 3, respectively, dated September 30, 2014 (ADAMS Accession No. ML14183B240).

The licensee has proposed several additional changes to TS 5.2, Organization, and TS 5.3, Facility Staff Qualification that were not included in Amendment Nos. 227 and 220. The first change is to capitalize the position of CERTIFIED FUEL HANDLER, consistent with its use as a defined term in TS 1.0, Definitions.

The NRC staff reviewed the proposed change to capitalize the position of CERTIFIED FUEL HANDLER where it is used in TS 5.2 and TS 5.3 and concludes the change is editorial in nature such that the current intent of the affected TS requirements is unchanged. Therefore, the staff finds that the licensee's proposed change to capitalize CERTIFIED FUEL HANDLER in TS 5.2 and TS 5.3, is acceptable.

The licensee has also proposed a change to Facility Staff TS 5.2.2.c (note that this was originally TS 5.2.2.b but was renumbered to TS 5.2.2.c by Amendment Nos. 227 and 220), to clarify that during unexpected absences of on-duty shift crew members, no fuel movement or movement of heavy loads over storage racks containing fuel is permitted. Specifically;

Facility Staff TS 5.2.2.c currently states:

> c. Shift crew composition may be less than the minimum requirement of Table 5.2.2-1 for a period of time not to exceed 2 hours in order to accommodate unexpected absence of on-duty shift crew members provided immediate action is taken to restore the shift crew composition to within the minimum requirements.

Revised Facility Staff TS 5.2.2.c would state:

> c. Shift crew composition may be less than the minimum requirement of Table 5.2.2-1 for a period of time not to exceed 2 hours in order to accommodate unexpected absence of on-duty shift crew members provided immediate action is taken to restore the shift crew composition to within the minimum requirements. During such absences, no fuel movement or movement of heavy loads over storage racks containing fuel is permitted.

The NRC staff has reviewed the proposed revision to TS 5.2.2.c restricting fuel movement or movement of heavy loads over storage racks containing fuel when an unexpected absence of the on-duty shift crew results in a minimum crew composition less than specified in TSs. The staff finds that additional restriction on fuel movement and heavy loads prudent considering the reduced staffing levels at a permanently shutdown and defueled reactor facility. Therefore, the NRC staff finds that the proposed licensee change to TS 5.2.2.c, is acceptable.

The licensee has also proposed a change to Facility Staff Qualifications TS 5.3.1 to delete the qualification requirements for multi-discipline supervisors. SCE states that it will no longer be utilizing the position of multi-discipline supervisor. Specifically;

Facility Staff Qualifications TS 5.3.1 currently states:

5.3.1 Each member of the facility staff shall meet or exceed the minimum qualifications of ANSI N18.1-1971 for comparable positions, except a) the radiation protection manager, who shall meet or exceed the qualifications of Regulatory Guide 1.8, September 1975, and b) multi-discipline supervisors who shall meet or exceed the qualifications listed below.

a. Education: Minimum of a high school diploma or equivalent.

b. Experience: Minimum of four years of related technical experience which shall include three years power plant experience of which one year is at a nuclear plant.

c. Training: Complete the multi-discipline supervisor training program.

Revised Facility Staff Qualifications TS 5.3.1 would state:

5.3.1 Each member of the facility staff shall meet or exceed the minimum qualifications of ANSI N18.1-1971 for comparable positions, except a) the radiation protection manager, who shall meet or exceed the qualifications of Regulatory Guide 1.8, September 1975.

The NRC Staff has reviewed the proposed deletion of the qualifications for multi-discipline supervisors from the SONGS Units 2 and 3 TSs and concluded that the qualifications are not necessary since the licensee no longer utilizes multi-discipline supervisors. Therefore, the NRC staff finds that the licensee's proposed change to TS 5.3.1, is acceptable.

3.7.17.1 Section 5.4, Technical Specification (TS) Bases Control

SONGS Units 2 and 3, TS 5.4, "Technical Specifications (TS) Bases Control," is a program that provides the requirements for changing the TS Bases without prior NRC approval. TS 5.4 will remain applicable with the reactor permanently shutdown and defueled. As such, it is being retained and revised, as follows, to reflect a permanently defueled condition.

Currently, the licensee is required to submit changes to the TS Bases to the NRC, which have been implemented without prior NRC approval, within 6 months following every Unit 3 refueling, not to exceed 24 months. The licensee has proposed to revise TS 5.4.4 to be consistent with the submittal of UFSAR updates for the permanently shutdown and defueled status of SONGS Units 2 and 3. The TS Bases changes (that do not require NRC approval) will be submitted to the NRC for information and/or review every 24 months consistent with the UFSAR updates.

The NRC staff has reviewed the proposed change to TS 5.4.4 that aligns the submittal of changes to the TS Bases to every 24 months consistent with the submittal of the UFSAR changes. The NRC staff has determined that the proposed revision to the frequency of submitting the TS Bases Control changes to NRC is administrative in nature. The revised TS

5.4.4 continues to meet the minimum frequency of the original TS. In addition, the change is consistent with the requirements of 10 CFR 50.71(e) for providing UFSAR updates to the NRC for a permanently shutdown and defueled reactor (i.e., every 24 months). Therefore, the NRC staff finds that the licensee's proposed change to TS 5.4.4, is acceptable.

3.7.17.2 Section 5.5.1, Procedures

TS 5.5.1, "Procedures," addresses procedures, programs and manuals required by the SONGS Units 2 and 3 TSs. The licensee proposes to delete that following procedures from the permanently defueled technical specifications:

TS 5.5.1.1, "Scope," requires that written procedures be established, implemented, and maintained covering certain activities. The licensee has proposed to delete TS 5.5.1.1, paragraphs b and f.

TS 5.5.1.1, paragraph b., currently states:

> The emergency operating procedures required to implement the requirements of NUREG-0737 and NUREG-0737, Supplement 1, as stated in Generic Letter 82-33;

NUREG-0737, "Clarification of TMI Action Plan Requirements," November 1980 (ADAMS Accession No. ML051400209), and NUREG-0737, Supplement 1, "Clarification of TMI Action Plan Requirements: Requirements for Emergency Response Capability," January 1983 (ADAMS Accession No. ML102560009), as stated in Generic Letter 82-33, "Supplement 1 to NUREG-0737 - Emergency Response Capabilities," dated December 17, 1982 (ADAMS Accession No. ML031080548), incorporated into one document all Three Mile Island (TMI)-related items approved for implementation by the Commission at that time. This included the use of human factored, function oriented, emergency operating procedures to improve human reliability and the ability to mitigate the consequences of a broad range of initiating events for operating reactors, and subsequent multiple failures or operator errors, without the need to diagnose specific events.

The licensee has proposed to delete the requirement of TS 5.5.1.1.b. because the emergency operating procedures discussed therein only pertain to accidents and events resulting from reactor operation. The licensee stated that the referenced procedures are no longer required for a permanently shutdown and defueled reactor.

The NRC staff reviewed the proposed deletion of TS 5.5.1.1.b. and determined that NUREG-0737, as supplemented, implemented programmatic changes to the way reactor operators are trained, instrumentation information is presented, and procedures are structured, using human factors and a function oriented approach to address operating events and accidents. These accidents, and the associated emergency operating procedures to detect, respond to, and mitigate such accidents, concerned malfunctions of the reactor and its supporting systems are not relevant to a permanently shutdown and defueled reactor, which is

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no longer authorized to operate or place fuel in the reactor vessel. Therefore, the NRC staff finds that the licensee's proposed change to delete TS 5.5.1.1.b., is acceptable.

TS 5.5.1.1.f., concerns the modification of the core protection calculator (CPC) addressable constants. Software modifications to constants, algorithms, or fuel cycle specific data shall be performed in accordance with the most recent version of "CPC Protection Algorithm Software Change Procedure," (CEN-39(A)-P). The licensee has proposed to delete TS 5.5.1.1.f. because the CPC is no longer required. The CPCs are one of two systems that monitor core power distribution online and derive the LHR and DNBR parameters and associated RPS trips. The TSs that rely on the CPC are TS 3.3.1 RPS Instrumentation - Operating and TS 3.3.3 Control Element Assembly Calculators, and are only applicable in MODES 1 and 2.

The NRC staff has determined that the instrumentation-related TS 3.3.1 and TS 3.3.3 that reference the CPC, as discussed in the Section 3.7.9 of this SE, are no longer required based on the permanent shutdown and defueled condition of SONGS Units 2 and 3. The CPC is part of the RPS to protect the reactor core from damage. Since SONGS Units 2 and 3 are not authorized to operate or emplace fuel in the reactor vessel, protection of the reactor core is no longer relevant, and a control procedure for the modification of the CPC, as required in TS 5.5.1.1.f., is unnecessary. Therefore, the NRC staff finds that the licensee's proposed change to delete TS 5.5.1.1.f., is acceptable.

3.7.17.3 Section 5.5.2, Programs and Manuals

TS 5.5.2.4, "Component Cyclic or Transient Limit Program," controls to track cyclic and transient occurrences to ensure that RCS components are monitored for fatigue evaluation based on a conservative estimate of the magnitude and frequency of the temperature and pressure transients resulting from normal operation, normal and abnormal load transients and accident conditions. The licensee proposes to delete this program since the RCS components monitored by this program are no longer used at SONGS Units 2 and 3 considering its permanently shutdown and defueled status.

The NRC staff has determined that deletion of the Component Cyclic or Transient Limit Program from TSs is consistent with the transition to a permanently shutdown and defueled facility. Since, in accordance with 10 CFR 50.82(a)(2), the licensee is prohibited from operating the plant or placing fuel in the reactor vessel, the RCS and reactor support systems are no longer in use. Consequently, the component cyclic or transient limit program is not relevant at SONGS Units 2 and 3 since the components monitored by the program are permanently out of service. Therefore, the NRC staff finds that the licensee's proposed change to delete TS 5.5.2.4, appropriately reflects the change in plant status, and is acceptable.

TS 5.5.2.5, "Reactor Coolant Pump Flywheel Inspection Program," provides for the inspection of the reactor coolant pump flywheels. The licensee proposed to delete this program since the reactor coolant pump flywheel is a component only used in support of reactor operation. Inspection of the reactor coolant pump flywheel is not relevant to SONGS Units 2 and 3 since the licensee is no longer authorized to operate the reactor or emplace fuel in the reactor vessel.

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The NRC staff has determined that deletion of the Reactor Coolant Pump Flywheel Inspection Program from TSs is consistent with the transition to a permanently shutdown and defueled facility. Since, in accordance with 10 CFR 50.82(a)(2), the licensee is prohibited from operating the plant or placing fuel in the reactor vessel, reactor coolant pumps are no longer used in support of any function at the facility. Therefore, the NRC staff finds that the licensee's proposed change to delete TS 5.5.2.5, appropriately reflects the change in plant status, and is acceptable.

TS 5.5.2.6, "Secondary Water Chemistry Program," provides controls for monitoring secondary water chemistry to inhibit SG tube degradation and low pressure turbine disc stress corrosion cracking. The licensee proposed to delete this program because the components that the program was established to protect, using water chemistry control, are associated with reactor operation. With the licensee's decision to cease reactor operations, these components are no longer in operation and do not need protection from degradation or stress corrosion cracking.

The NRC staff has determined that the deletion of the Secondary Water Chemistry Program is consistent with the transition to a permanently shutdown and defueled facility. Since, in accordance with 10 CFR 50.82(a)(2), the licensee is prohibited from operating the plant or placing fuel in the reactor vessel, the SGs and turbine are no longer used in support of any function at the facility. Therefore, the NRC staff finds that the licensee's proposed change to delete TS 5.5.2.6, appropriately reflects the change in plant status, and is acceptable.

TS 5.5.2.7, "Explosive Gas and Storage Tank Radioactivity Monitoring Program," provides controls for potentially explosive gas mixtures in the gaseous radwaste system, and the quantity of radioactivity contained in unprotected outdoor liquid storage tanks. The licensee has proposed to revise the explosive gas and storage tank radioactivity monitoring program to be consistent with the permanently shutdown and defueled condition of the SONGS Units 2 and 3 facility. Paragraphs a. and b. of the program are being deleted because these portions of the explosive gas and storage tank radioactivity monitoring program pertain only to reactor support systems that are no longer needed due to SONGS permanently shutdown and defueled condition. Specifically, there will no longer be any source of explosive or radioactive gases generated from reactor operation. In addition, the licensee states that the gaseous radwaste system and the waste gas decay tank have been vented and removed from service. As such, references to potentially explosive gas mixtures and methods for determining gaseous radioactivity have been deleted. The licensee has proposed to retain the storage tank radioactivity monitoring program as modified below:

TS 5.5.2.7 <u>Storage Tank Radioactivity Monitoring Program</u>

This program provides controls for the quantity of radioactivity contained in unprotected outdoor liquid storage tanks. The liquid radwaste quantities shall be determined in accordance with the

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Standard Review Plan, Section 15.7.3, "Postulated Radioactive Release due to Tank Failures."

The program shall include a surveillance program to ensure that the quantity of radioactivity contained in all outdoor liquid radwaste tanks that are not surrounded by liners, dikes, or walls, capable of holding the tanks' contents and do not have tank overflows and surrounding area drains connected to the Liquid Waste Management System is less than the amount that would result in concentrations less than the limits of 10 CFR Part 20, Appendix B, Table II, Column 2, at the nearest potable water supply and the nearest surface water supply in an unrestricted area, in the event of an uncontrolled release of the tanks' contents.

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Storage Tank Radioactivity Monitoring Program surveillance frequencies.

The NRC staff has reviewed the proposed revision to the Storage Tank Radioactivity Monitoring Program. The staff finds the proposed changes prudent given the uncertainty in how future radwaste generated by flushing and cutting of radioactive systems will be stored and processed. Therefore, the NRC staff finds that the licensee's proposed change to TS 5.5.2.7, Storage Tank Radioactivity Monitoring Program, is acceptable.

TS 5.5.2.8, "Primary Coolant Sources Outside Containment Program," was established to minimize leakage from portions of systems outside containment that could contain highly radioactive fluids during a serious transient or accident. The licensee proposed to delete this program since primary coolant systems have been drained at SONGS Units 2 and 3 and there are no longer any transient or accident conditions associated with primary coolant sources given the permanently shutdown and defueled condition of the plant.

The NRC staff has determined that deletion of TS 5.5.2.8, "Primary Coolant Sources Outside Containment Program," is consistent with the transition to a permanently shutdown and defueled facility. Since the licensee has certified its permanent cessation of operations and defueling in accordance with 10 CFR 50.82(a)(2), the licensee is prohibited from operating the reactors or placing fuel in the reactor vessels. Consequently, there are no DBAs involving reactor operation or refueling and there can no longer be any transients or accidents involving primary coolant outside of containment. Therefore, the NRC staff finds that the licensee's proposed change to delete TS 5.5.2.8, appropriately reflects the change in plant status, and is acceptable.

TS 5.5.2.9, "Pre-Stressed Concrete Containment Tendon Surveillance Program," provides controls for monitoring any tendon degradation in the pre-stressed concrete containment. The licensee has proposed to delete this program because the status of the containment is not relevant to the permanently shutdown and defueled reactors at SONGS Units 2 and 3.

The NRC staff considers that TS 5.5.2.9, "Pre-Stressed Concrete Containment Tendon Surveillance Program," is only applicable to a reactor authorized to operate or retain irradiated fuel in the reactor vessel. Pursuant to the licensee's certifications under 10 CFR 50.82(a)(2), the license is prohibited from operating the reactors or placing fuel in the reactor vessels at SONGS Units 2 and 3. Consequently, there are no DBAs involving reactor operation or refueling and no reliance on the containment to mitigate operating reactor DBAs. Thus, the staff has determined that containment tendon surveillance program TS is no longer applicable. Therefore, the NRC staff finds that the licensee's proposed change to delete TS 5.5.2.9, is acceptable.

TS 5.5.2.10, "Inservice Inspection and Testing Program," establishes the controls for periodic inspection and testing of the American Society of Mechanical Engineers (ASME) Code Class 1, 2, and 3 pumps and valves in accordance with the ASME Operation and Maintenance Code. These code classes protect equipment relied upon to prevent and mitigate DBAs. The licensee proposed to delete this program since there is no longer any ASME Code Class 1, 2 or 3 pumps and valves, or Code Class CC or MC components in the SONGS Units 2 and 3 inservice inspection and testing program that continue to operate and perform a specific function in mitigating the consequences of a reactor accident due to the permanently shutdown and defueled status of the plants.

Because the licensee is prohibited from operating the plant or placing fuel in the reactor vessel, in accordance with 10 CFR 50.82(a)(2), there are no longer any ASME Code class pumps and valves that remain in operation and are to be relied upon to mitigate a DBA. As such, the inservice inspection and testing program is no longer relevant to SONGS Units 2 and 3, given the permanently shutdown and defueled status of these facilities. The NRC staff also notes that the licensee shall continue to monitor the performance and condition of all SSCs associated with the storage, control, or maintenance of spent fuel in in a safe condition and with reasonable assurance that these SSCs are capable of fulfilling their intended functions, pursuant to 10 CFR 50.65, ""Requirements for monitoring the effectiveness of maintenance at nuclear power plants." Therefore, the NRC staff finds that the licensee's proposed change to delete TS 5.5.2.10, Inservice Inspection and Testing Program, appropriately reflects the change in plant status, and is acceptable.

TS 5.5.2.11, "Steam Generator (SG) Program," ensures that the SG tube integrity is maintained. The licensee proposed to delete this program since SONGS Units 2 and 3 are permanently defueled and not authorized to operate; therefore, the SGs are no longer functional and the SG tubes will not be subjected to the temperature and pressure effects that the SG program was put in place to protect against.

The NRC staff has determined that the SG program is only relevant to an operating reactor where the SGs are used for removing heat associated with reactor operation. Since the licensee has certified its permanent cessation of operations and defueling in accordance with 10 CFR 50.82(a)(2), the licensee is prohibited from operating the reactors or placing fuel in the reactor vessels. Consequently, the SGs are no longer used in support of any function at the facility. Therefore, the NRC staff finds that the licensee's proposed change to delete TS 5.5.2.11, appropriately reflects the change in plant status, and is acceptable.

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TS 5.5.2.12, "Ventilation Filter Testing Program (VFTP)," establishes the required testing and frequency of the CREACUS high efficiency particulate filters and charcoal adsorbers utilized by the system.

The VFTP is being deleted because it pertains only to reactor support systems that does not apply in a permanently defueled condition. As noted, in part, by the licensee in its license amendment request, dated March 21, 2014, "[t]he accident analysis applicable to the permanently defueled condition does not rely on ventilation filters for accident mitigation."

The NRC staff has determined that reference to the VFTP only appears in SONGS Units 2 and 3 TSs in three places: TS 5.5.2.12; TS 3.7.11 "Control Room Emergency Air Cleanup System (CREACUS)" (SR 3.7.11.2 and SR 3.7.11.4); and TS 5.5.2.16.d of the "Control Room Envelope Habitability Program." The VFTP is used to confirm the function and operability of the CREACUS. The NRC staff has evaluated CREACUS in Section 3.7.13 (TS 3.7.11) and found that CREACUS is no longer required in the SONGS TSs per Criterion 3 of 10 CFR 50.36(c)(2)(ii)(C). Since TS 5.5.2.12 "Ventilation Filter Testing Program" only exists to support the SRs of TS 3.7.11 (i.e. SR 3.7.11.2 and SR 3.7.11.4, respectively) and since the NRC approves that deletion of TS 3.7.11, the NRC staff finds the licensee's proposed change to delete TS 5.5.2.12, is acceptable.

TS 5.5.2.13, "Diesel Fuel Oil Testing Program," pertains to the testing of both new and stored fuel oil used to supply the EDGs. The accident analyses applicable to the permanently shutdown and defueled condition at SONGS no longer rely on EDGs for accident mitigation. The requirement for EDGs, which are supported by the fuel oil being tested per this program, has been proposed for deletion from the TSs.

The NRC staff has reviewed the proposed changes against the requirements in 10 CFR 50.36 and Chapter 15 of the SONGS UFSAR and concluded that the EDG fuel oil and lube oil system are not required. These support systems to the EDGs are not required because there are no active systems or associated support systems credited as part of the initial conditions of an analysis or as part of the primary success path for mitigation of the FHA DBA. The staff confirmed that there are no other DBAs that rely on EDGs or the EDG support systems. In addition, the NRC staff approves the deletion of TS 3.8.3, "Diesel Fuel Oil, Lube Oil, and Starting Air," in Section 3.7.14 of this SE. Therefore, the NRC staff finds that the licensee's proposed change to delete TS 5.5.2.13, the diesel fuel oil testing program, is acceptable.

TS 5.5.2.15, "Containment Leakage Rate Testing Program," is being proposed for deletion because the containment leakage rate testing program pertains only to verifying the operability of the containment systems. The need for containment or the associated required TSs does not apply in a permanently shutdown and defueled condition. The requirements for containment systems (i.e. TS 3.6.1, TS 3.6.2, TS 3.6.3, TS 3.6.4, TS 3.6.5, TS 3.6.6.1, TS 3.6.6.2, TS 3.6.8 and TS 3.9.3) are being deleted, as described in Section 3.7.12 of this SE.

Primary containment integrity and isolation are only required for post-accident conditions from power operations. However, 10 CFR 50.82(a)(2) prohibits the licensee from operating the plant

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or placing fuel in the reactor vessel. Therefore TS 3.6.1, TS 3.6.2, TS 3.6.3, TS 3.6.4, TS 3.6.5, TS 3.6.6.1, TS 3.6.6.2, TS 3.6.8 and TS 3.9.3, which address primary containment integrity and isolation during power operations and refueling operations, are no longer applicable. The program specified TS 5.5.2.15 requires the implementation of containment leakage rate testing in accordance with 10 CFR Part 50 Appendix J, Option B, "Performance-Based Requirements." The TS 5.5.2.15 program is no longer needed since 10 CFR 50.54(o) excludes permanently defueled units from the requirements of 10 CFR Part 50 Appendix J, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors." Therefore, the NRC staff finds that the licensee's proposed change to delete TS 5.5.2.15, Containment Leakage Rate Testing Program, is acceptable.

TS 5.5.2.16, "Control Room Envelope Habitability Program," ensures that adequate radiation protection is provided to permit access and occupancy of the CRE under DBA conditions without personnel receiving radiations exposures above limits. The licensee has proposed this program for deletion because the CRE is not required for providing airborne radiological protection for the control room operators for the remaining DBAs at SONGS Units 2 and 3 based on the permanently shutdown and defueled status of the facility.

The NRC staff evaluated the remaining accident analyses at SONGS Units 2 and 3 and confirmed that no ESF system is credited in the mitigation of the CR, EAB, or LPZ dose consequences, as detailed in Sections 3.2 through 3.6 of this SE. This includes no credit for the FHIS, the fuel handling building PACU filtration system, the CRIS and the CREACUS. The evaluation of the DBAs applicable to the permanently shutdown and defueled facility demonstrate that the dose consequences within the CRE are acceptable without relying on SSCs remaining functional for accident mitigation, including FHAs. (The one exception to this is the continued function of the passive fuel storage pool structure, which will be maintained as a TS for SONGS.)

Reference to the "Control Room Envelope Habitability Program" only appears in the current SONGS Units 2 and 3 TSs in two places: TS 5.5.2.16, "Control Room Envelope Habitability Program" and TS 3.7.11, "Control Room Emergency Air Cleanup System (CREACUS)" (SR 3.7.11.4).

The NRC staff previously determined in its evaluation of TS 3.7.11, "CREACUS," Section 3.7.13 of this SE, that CREACUS no longer satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii)(C). Consequently, the NRC staff has approved the deletion of TS 3.7.11 for SONGS Units 2 and 3. Since the Control Room Envelope Habitability Program only exists to support a surveillance requirement of TS 3.7.11 (i.e. SR 3.7.11.4) and since TS 3.7.11 will be deleted, the NRC staff finds that the licensee's proposed change to delete TS 5.5.2.16, Control Room Envelope Habitability Program, is acceptable.

TS 5.5.2.17, "Battery Monitoring and Maintenance Program," provides controls for safety-related battery maintenance. The licensee proposes deletion of this program consistent with the deletion of the corresponding TS for DC electrical systems and associated batteries. The licensee states that the SONGS accident analyses do not rely on batteries for any accident mitigation.

The NRC staff has reviewed the proposed changes against the requirements in 10 CFR 50.36 and Chapter 15 of the SONGS UFSAR and concluded that the DC electrical distribution system batteries are not required. The support systems to the DC electrical distribution system, including the batteries, are not required because there are no active systems or associated support systems credited as part of the initial conditions of an analysis or as part of the primary success path for mitigation of the remaining DBAs at SONGS Units 2 and 3. In addition, the NRC staff has approved the deletion of TS 3.8.6, "Battery Parameters," in Section 3.7.14 of this SE. Therefore, the NRC staff finds that the licensee's proposed change to delete the TS 5.5.2.17, the battery monitoring and maintenance program, is acceptable.

3.7.17.4 Section 5.6, Safety Function Determination Program (SFDP)

The SONGS Units 2 and 3, "Safety Function Determination Program (SFDP)," as detailed in TS 5.6.1, TS 5.6.2, TS 5.6.3 and TS 5.6.4, ensures that a loss of safety function is detected and appropriate actions taken. Upon failure to meet two or more LCOs at the same time, an evaluation shall be made to determine if loss of safety function exists. The program implements the requirements of LCO 3.0.6. LCO 3.0.6 directs an evaluation in accordance with the SFDP to determine if a loss of safety function exists based on the status of redundant TS safety systems and associated support systems (systems that support the functionality of the safety system) to ensure the appropriate required actions are taken to maintain overall reactor safety. There are no active SSCs at SONGS Units 2 and 3 that are required for accident mitigation with the permanent cessation of reactor operations and the permanent removal of the fuel from the reactor vessels, as discussed in the evaluation of the remaining DBAs in Sections 3.2 through 3.6 of this SE. Therefore, the requirements of the SFDP, which directs cross-train checks of multiple and redundant safety systems, no longer apply.

Based on the permanently shutdown and defueled status of SONGS Units 2 and 3, all specifications for the active systems from the defueled TSs have been proposed for deletion by this licensing action. Consequently, the SFDP is no longer meaningful. In addition, the SFDP is invoked by LCO 3.0.6, which is being deleted in its entirety, as discussed in Section 3.7.6 of this SE. Therefore, the NRC staff finds that the licensee's proposed change to delete TS Section 5.6, Safety Function Determination Program, is acceptable.

3.7.17.5 Section 5.7, Reporting Requirements

TS 5.7.1.1, "Annual Reports," requires a Reactor Coolant System Specific Activity Report in accordance with TS 5.7.1.1.b. The report gathered data on reactor conditions when the I-131 or gross specific activity of the reactor coolant exceeded limits specified in TS 3.4.16. The licensee has proposed to delete SONGS Units 2 and 3, TS 5.7.1.1.b, "Reactor Coolant System Specific Activity Report," since it is not applicable to a permanently shutdown and defueled reactor.

The NRC staff has reviewed the proposed deletion of TS 5.7.1.1 concerning the Reactor Coolant Specific Activity Report. The facility RCSs have been drained and the activity of the RCS is no longer relevant to the SONGS Units 2 and 3 in their permanently shutdown and

defueled status. In addition, as noted above, TS 5.7.1.1.b only exists to analyze data related to the exceedance of limits specified in TS 3.4.16. Since RCS activity is not meaningful for SONGS and TS 3.4.16 will be deleted, the NRC staff finds that the licensee's proposed change to delete TS 5.7.1.1, is acceptable.

TS 5.7.1.2, "Annual Radiological Environmental Operating Report," covers summaries, interpretations, and analyses of trends related to the radiological environmental monitoring program, for each unit, during the previous calendar year.

The licensee has proposed to revise the TS description by replacing applicability of the report to the "facility" rather than to each 'unit." In addition, the licensee is deleting a Note indicating "a single submittal may be made for a multiple unit station." This note is no longer necessary since the SONGS facility is no longer treated as a multiunit site for the purposes of the annual radiological environmental operating report.

The NRC staff reviewed the proposed revision to TS 5.7.1.2 and concludes that changing the word "unit" to "facility" and the deletion of the multiple unit station note is a clarifying change that is editorial in nature such that the current intent of the requirement is unchanged. Therefore, the NRC staff finds that the licensee's proposed change to TS 5.7.1.2, Annual Radiological Environmental Operating Report, is acceptable.

TS 5.7.1.3, "Radioactive Effluent Release Report," covers "...the operation of the unit during the previous calendar year...." In addition, the report shall summarize the "...effluents released from the unit," and "... radioactive waste shipped from the unit directly..." and "... radioactive waste shipped from the unit's intermediary processor..."

The licensee proposed to revise the TS description by replacing "unit" with "facility" such that the description will state "... the operation of the facility during the previous calendar year ..." and, effluents "... released from the facility" and "... radioactive waste shipped from the facility directly ..." and "... radioactive waste shipped from the facility's intermediary processor...." In addition, the licensee is deleting a Note indicating "a single submittal may be made for a multiple unit station." This note is no longer necessary since the SONGS facility is no longer treated as a multiunit site for the purposes of the radioactive effluent release report.

The NRC staff reviewed the proposed revision to TS 5.7.1.3 and concludes that changing the word "unit" to facility" and the deletion of the multiple unit station note is a clarifying change that is editorial in nature such that the current intent of the requirement is unchanged. Therefore, the NRC staff finds that the licensee's proposed change to TS 5.7.1.3, Radioactive Effluent Release Report, is acceptable.

TS 5.7.1.5, "Core Operating Limits Report (COLR)," establishes the core operating limits prior to each reload cycle. The licensee proposed to delete this program since it is prohibited from reloading fuel into the SONGS Units 2 and 3 reactor core and the safety limits established by this report no longer apply.

The NRC staff has determined that the proposed deletion of the COLR would appropriately reflect the permanently shutdown and defueled condition of the facility. The COLR only applies to reactors authorized to operate. Since the licensee is prohibited from operating the SONGS reactors or placing fuel in the reactor vessels, the COLR is no longer necessary. Therefore, the NRC staff finds that the licensee's proposed change to delete TS 5.7.1.5, is acceptable.

TS 5.7.1.6, "Reactor Coolant System (RCS) Pressure and Temperature Limits Report (PTLR)," documents the pressure and temperature limits for heatup, cooldown, heatup and cooldown rates, low temperature operation, criticality, and hydrostatic testing as referenced in the following TSs:

- TS 3.4.3 RCS Pressure and Temperature (P/T) Limits
- TS 3.4.6 RCS Loops – Mode 4
- TS 3.4.7 RCS Loops – Mode 5, Loops Filled
- TS 3.4.12.1 Low Temperature Overpressure Protection (LTOP) System, RCS Temperature≤PTLR Limit
- TS 3.4.12.2 Low Temperature Overpressure Protection (LTOP) System, RCS Temperature>PTLR Limit

The licensee proposes to delete this program since the RCS is no longer used at SONGS Units 2 and 3 in its permanently shutdown and defueled status.

The NRC staff has determined that deletion of the Reactor Coolant System Pressure and Temperature Limits Report from TSs is consistent with the transition to a permanently shutdown and defueled facility. Since, in accordance with 10 CFR 50.82(a)(2), the licensee is prohibited from operating the reactors or placing fuel in the reactor vessels, the RCS and reactor support systems are no longer in use. Consequently, the RCS PTLR is not relevant at SONGS Units 2 and 3 since the RCS is no longer functional. The staff notes that the change is consistent with the deletion of the Section 3.4 RCS TSs that reference the PTLR as discussed in Section 3.7.10 of this SE. Therefore, the NRC staff finds that the licensee's proposed change to delete TS 5.7.1.6, appropriately reflects the change in the SONGS plant status, and is acceptable.

TS 5.7.1.7, "Hazardous Cargo Traffic Report," requires that SCE monitors the hazardous cargo traffic on Interstate Highway 5 and the railroad line near SONGS and submits the results to the NRC Regional Administrator once every 3 years. This reporting requirement addressed potential changes in use characteristics of these transportation routes over the life of the facility. In the enclosure to the license amendment request dated March 21, 2014, SCE proposed to delete this reporting requirement from the TSs. In the supplement dated February 23, 2015, SCE stated that it would continue to perform the hazardous traffic report in accordance with a licensee-controlled documents.

The requirements of 10 CFR 50.36(c)(5) state that Administrative Controls TSs should include reporting necessary to assure operation of the facility in a safe manner. The reporting requirements included in Section 5.6 of NUREG-1432, "Standard Technical Specifications – Combustion Engineering Plants," Volume 1 (ADAMS Accession No. ML12102A165), include only those reports specified in the LCOs and those required by regulation. The Hazardous

Cargo Traffic Report does not directly relate to operation of the facility in a safe manner. Rather, it helps identify changes in the site environs that should be periodically assessed to ensure that the scope of events considered in the design-basis remains adequate. Consequently, the report does not significantly contribute to assuring operation in a safe manner. Therefore, the NRC staff finds that the licensee's proposed change to delete TS 5.7.1.7, Hazardous Cargo Traffic Report, and implement a similar reporting requirement in a licensee controlled document, is acceptable.

TS 5.7.2, "Special Reports," provides a description and requirements regarding reports related to inspections, tests, and maintenance activities as directed in other SONGS TSs. The listed Special Reports pertain to 1) a pre-planned alternate method of monitoring post-accident instrumentation functions, 2) abnormal degradation of the containment structure detected during tests required by the Pre-Stressed Concrete Containment Tendon Surveillance Program, and 3) a report, following entry into MODE 4, concerning inspections performed in accordance with the SG program. The licensee states that these reports are being deleted because they do not apply in a permanently defueled condition.

The NRC staff concludes that the TS required special report information on inspections, tests, and maintenance activities for safety-related instrumentation, containment, and SGs, apply to SSCs that are no longer relevant at a permanently shutdown and defueled SONGS reactors. In addition, the NRC has approved the deletion of the associated TSs for the SSC that are subject to these special reports from the SONGS Unit 2 and 3 permanently defueled TSs. Specifically:

(1) the special report for a pre-planned alternate method of monitoring post-accident instrumentation functions is no longer necessary since the post-accident monitoring instrumentation in TS 3.3.11 is being deleted from the SONGS defueled TSs, as discussed in Section 3.7.9 of this SE.

(2) the special report on abnormal degradation of the containment structure detected during tests required by the Pre-Stressed Concrete Containment Tendon Surveillance Program is no longer necessary since the tendon surveillance program in TS 5.5.2.9 is being deleted from the SONGS defueled TSs, as discussed in Section 3.7.17.3 of this SE.

(3) the special report, following entry into MODE 4, concerning inspections performed in accordance with the SG program is no longer necessary since the SG program in TS 5.5.2.11 is being deleted from the SONGS defueled TSs, as discussed in Section 3.7.17.3 of this SE.

Therefore, the NRC staff finds that the licensee's proposed change to delete TS 5.7.2, Special Reports, is acceptable.

3.8 Changes to Facility Operating License

In SCE's March 21, 2014, license amendment request, as supplemented by a letters dated February 25, 2015, and March 18, 2015, the licensee proposed to remove, modify, and add, several facility operating license conditions, based on the permanently shutdown and defueled status of SONGS Units 2 and 3.

3.8.1 Changes to License Condition 2.B.(2)

Currently License Condition 2.B.(2), for SONGS Units 2 and 3, reads:

> (2) Southern California Edison Company (SCE), pursuant to Section 103 of the Act and 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," to possess, use, and operate the facility at the designated location in San Diego County, California, in accordance with the procedures and limitations set forth in this license.

The licensee is proposing to strike reference in the license condition to "...operate..." the facility.

The revised License Condition 2.B.(2) will read, as follows:

> (2) Southern California Edison Company (SCE), pursuant to Section 103 of the Act and 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," to possess and use the facility at the designated location in San Diego County, California, in accordance with the procedures and limitations set forth in this license.

Pursuant to 10 CFR 50.82(a)(2), as a result of the 10 CFR 50.82(a)(1)(i) and 10 CFR 50.82(a)(1)(ii) certifications submitted by the licensee, the 10 CFR Part 50 licenses for SONGS Units 2 and 3 no longer authorize operation of the reactors. As such, reference to operation of the facility in License Condition 2.B.(2) is inconsistent with the limitation imposed on the licensee by 10 CFR 50.82(a)(2). Therefore, the NRC staff finds the licensee's proposed change to License Condition 2.B.(2) provides consistency with 10 CFR 50.82(a)(2) and, is acceptable.

3.8.2 Changes to License Condition 2.B.(3)

Currently License Condition 2.B.(3), for SONGS Units 2 and 3, reads:

> (3) SCE, pursuant to the Act and 10 CFR Part 70, to receive, possess, and use at any time special nuclear material as reactor fuel, in accordance with the limitations for storage and amounts required for reactor operation, as described in the Final Safety Analysis Report, as supplemented and amended;

The licensee is proposing to revise this license condition to read, as follows:

> (3) SCE, pursuant to the Act and 10 CFR Part 70, to possess at any time special nuclear material that was used as reactor fuel, in accordance with the limitations for storage, as described in the Final Safety Analysis Report, as supplemented and amended;

The licensee states the proposed revision to this license condition is consistent with the restrictions of 10 CFR 50.82(a)(2) that no longer authorizes operation or emplacement of fuel in the reactor vessels at SONGS Units 2 and 3.

The proposed change removes the authorization for receipt and use of special nuclear material (SNM) as reactor fuel and eliminates the reference to use of the SNM for reactor operations. The proposed change also limits the possession of SNM pursuant to the license condition as being "that was used" as reactor fuel. Pursuant to 10 CFR 50.82(a)(2) the 10 CFR Part 50 licenses for SONGS Units 2 and 3 no longer authorize operation of the reactors. As such, the licensee has no need to receive SNM in the form of reactor fuel and cannot use SNM as reactor fuel for reactor operations. The continued authorization to possess SNM "that was used" as reactor fuel is necessary as the licensee currently possesses the reactor fuel that was used for the past operations of the reactor. Based on the above, the NRC staff finds the licensee's proposed change to License Condition 2.B.(3) is consistent with the permanently shutdown status of SONGS Units 2 and 3 and is, therefore, acceptable.

3.8.3 Changes to License Condition 2.B.(4)

Currently License Condition 2.B.(4), for SONGS Units 2 and 3, reads:

> (4) SCE, pursuant to the Act and 10 CFR Parts 30, 40, and 70, to receive, possess, and use at any time any byproduct, source and special nuclear material as sealed neutron sources for reactor startup, sealed sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required;

The licensee is proposing to revise this license condition to read, as follows:

> (4) SCE, pursuant to the Act and 10 CFR Parts 30, 40, and 70, to receive, possess, and use at any time any byproduct, source and special nuclear material as sealed sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required; and possess any byproduct, source and special material as sealed neutron sources that was used for reactor startup;

The licensee states the proposed revision to this license condition is consistent with the restrictions of 10 CFR 50.82(a)(2) that no longer authorizes operation or emplacement of fuel in the reactor vessels at SONGS Units 2 and 3. The proposed changes remove the authorization for receipt and use of byproduct, source, and SNM as sealed neutron sources for reactor startup but retains authorization to possess such sources previously used for reactor startup. The deletion of the authorization to receive and use sources for reactor startup is consistent with the fact that SONGS Units 2 and 3 are no longer authorized to operate and the continued authorization to possess neutron sources that were used for reactor startup is consistent with the safe storage of byproduct, source, and SNM. As such, the NRC staff finds that the licensee's proposed change to License Condition 2.B.(4), is consistent with the permanently shutdown status of the facilities and is, therefore, acceptable.

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3.8.4 Changes to License Condition 2.C.(1)

Current License Condition 2.C.(1), for SONGS Units 2 and 3, reads:

Maximum Power Level

(1) Southern California Edison Company (SCE) is authorized to operate the facility at reactor core power levels not in excess of full power (3438 megawatts thermal).

The licensee is proposing to delete this license condition, which will read:

(1) Deleted

The licensee states that this license condition can be deleted because SONGS Units 2 and 3 are permanently shut down and defueled in accordance with 10 CFR 50.82(a)(2) and therefore power operation is no longer authorized.

The NRC staff has reviewed the proposed deletion of License Condition 2.C.(1) and determined that power operation is no longer authorized at SONGS Units 2 and 3 based on the licensee's 10 CFR 50.82(a)(2) certifications of being permanently shutdown and defueled. The licensee is not authorized to operate the SONGS Units 2 and 3 at any power. Therefore, the NRC staff finds the licensee's proposed change to delete License Condition 2.C.(1) is appropriate and, is acceptable.

3.8.5 Changes to License Condition 2.C.(14) [Unit 2] and License Condition 2.C.(12) [Unit 3]

Current License Condition 2.C.(14) for SONGS Units 2 and License Condition 2.C.(12) for SONGS Unit 3, read:

Fire Protection (Section 9.5.1, SER, SSER #4, SSER #5, Section 1.12, SSER #5; SE dated November 15, 1982; Revision 1 to Updated Fire Hazards Analysis Evaluation dated June 29, 1988)

SCE shall implement and maintain in effect all provisions of the approved fire protection program. This program shall be (1) as described in the Updated Fire Hazards Analysis through Revision 3 as revised by letters to the NRC dated May 31, July 22. and November 20, 1987 and January 21, February 22, and April 21, 1988; and (2) as approved in the NRC staff's Safety Evaluation Report (SER) (NUREG-0712) dated February 1981; Supplements 4 and 5 to the SER, dated January 1982 and February 1982, respectively; and the safety evaluation dated November 15, 1982; as supplemented and amended by the Updated Fire Hazards Analysis Evaluation for San Onofre 2 and 3, Revision 1 dated June 29, 1988. SCE may make changes to the approved fire protection program without prior approval of the Commission

only if those changes would not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire.

The licensee is proposing to delete License Condition 2.C.(14) for SONGS Unit 2 and delete License Condition 2.C.(12) for SONGS Unit 3, which will read:

Unit 2

(14) Deleted

Unit 3

(12) Deleted

The licensee states that this license condition is based on maintaining an operational fire protection program in accordance with 10 CFR 50.48, "Fire protection," with the ability to achieve and maintain safe shutdown of the reactor in the event of a fire and is no longer applicable at SONGS Units 2 and 3. However, many of the elements that are applicable for the operating plant fire protection program continue to be applicable during plant decommissioning. During the decommissioning process, a fire protection program is required by 10 CFR 50.48(f) to address the potential for fires that could result in a radiological hazard. However, the regulation is applicable regardless of whether a requirement for a fire protection program is included in the facility license. Therefore, a license condition requiring such a program for a permanently shutdown and defueled plant is not needed.

The NRC staff finds that License Conditions 2.C.(14) and 2.C.(12), "Fire Protection," for SONGS Units 2 and 3, respectively, are based on maintaining fire protection programs that provides reasonable assurance that the ability to achieve and maintain safe shutdown in the event of a fire in accordance with 10 CFR 50.48. Achieving and maintaining safe shutdown in the event of a fire is no longer applicable to the decommissioned fire protection programs at SONGS Units 2 and 3, since units are permanently shutdown and the fuel has been removed from the reactors. However, elements of the fire protection program continue during decommissioning to address fire events that could result in radiological hazards. The regulation in 10 CFR 50.48(f) requires SONGS Units 2 and 3 to address the potential for fires, which could result in a radiological hazard. The licensee has proposed that the rule is sufficient to ensure that a program is maintained and therefore having a license condition that also requires fire protection programs for the permanently shutdown and defueled units is redundant. Basis on the evaluation above, the NRC staff concludes that reliance on 10 CFR 50.48(f) is appropriate and the fire protection license condition is no longer necessary. Therefore, the NRC staff finds that the licensee's proposed change to delete License Condition 2.C.(14) for SONGS Units 2, and License Condition 2.C.(12) for SONGS Unit 3, is acceptable.

3.8.6 Changes to License Condition 2.C.(27) [Unit 2] and License Condition 2.C.(28) [Unit 3]

Current License Condition 2.C.(27) for SONGS Unit 2, and License Condition 2.C.(28) for SONGS Unit 3, read:

> Upon implementation of Amendment No. 214 [Unit 2 and Amendment No. 206, Unit 3] adopting TSTF 448, Revision 3, the determination of control room envelope (CRE) unfiltered air inleakage as required by SR 3.7.11.4 in accordance with TS 5.5.2.16.c(i), the assessment of CRE habitability as required by Specification 5.5.2.16.c(ii), and the measurement of CRE pressure as required by Specification 5.5.2.16.d, shall be considered met. Following implementation:
>
> (a) The first performance of SR 3.7.11.4, in accordance with Specification 5.5.2.16.c(i) shall be within the specified frequency of 6 years, plus the 18-month allowance of SR 3.0.2, as measured from May 18, 2004, the date of the most recent successful tracer gas test, as stated in the September 17, 2004 letter response to Generic Letter 2003-01, or within the next 18 months if the time period since the most recent successful tracer gas test is greater than 6 years.
>
> (b) The first performance of the periodic assessment of CRE habitability, Specification 5.5.2.16.c(ii), shall be within 3 years, plus the 9-month allowance of SR 3.0.2, as measured from May 18, 2004, the date of the most recent successful tracer gas test, as stated in the September 17, 2004, letter response to Generic Letter 2003-01, or within the next 9 month if the time period since the most recent successful tracer gas is greater than 3 years.
>
> (c) The first performance of the periodic measurement of CRE pressure, Specification 5.5.2.16.d, shall be within 6 months.

The licensee is proposing to delete License Condition 2.C.(27) for SONGS Unit 2 and License Condition 2.C.(28) for SONGS Unit 3, which will read:

<u>Unit 2</u>

(27) Deleted

<u>Unit 3</u>

(28) Deleted

The NRC staff evaluated the remaining accident analyses at SONGS Units 2 and 3 and confirmed that no ESF system is used to mitigate the CR, EAB, or LPZ dose consequences, as

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detailed in Sections 3.2 through 3.6 of this SE. This includes no credit for the FHIS, the fuel handling building PACU filtration system, the CRIS and the CREACUS. Since SONGS Units 2 and 3 are permanently shut down and defueled, and greater than 17 months of decay time has elapsed since permanent shut down, the remaining DBAs applicable to the facility demonstrate that the dose consequences within the CRE are acceptable without relying on SSCs remaining functional for accident mitigation, with the exception of the passive fuel storage pool structure. In addition, the staff has determined that related CREACUS TS 3.7.11, and the Control Room Envelope Habitability Program TS 5.5.2.16, are no longer needed, as discussed in Sections 3.7.13 and 3.7.17.3, respectively, of this SE. Based on the discussion above, the NRC staff finds that the licensee's proposed change to delete of SONGS Unit 2 License Condition 2.C.(27), and SONGS Unit 3 License Condition 2.C.(28), is acceptable.

3.8.7 New License Condition 2.C.(28) [Unit 2] and License Condition 2.C.(29) [Unit 3]

By letter dated February 25, 2015 (ADAMS Accession No. ML15058A033), the licensee responded to an RAI from the NRC staff regarding the actions that will be taken by SCE to provide reasonable assurance that the passive, long-lived structures and components in the SFP, the fire protection system, and the radiation protection system, will be maintained in a safe condition beyond the normal licensed operating period of 40 years, pursuant to the provisions of 10 CFR 50.51(b). The NRC staff asked the licensee to identify and list the long-live, passive structures and components. In addition, the staff requested a summary of actions that will be taken to monitor and maintain the long-lived, passive structures and components. One of the staff's concerns involved the aging of neutron absorbing materials used for criticality control in SFPs.

SCE responded to the specific concern on the use of neutron absorbing materials in the SFP racks at SONGS. SCE noted that the SONGS SFP racks do contain Boraflex, a neutron-absorbing material. However, no credit is taken in SONGS accident analyses or licensing basis for the existence of the Boraflex. In addition, the NRC previously evaluated and approved borated stainless steel rods that may be placed in fuel assembly guide tubes (GTs) for reactivity control. This feature has not been implemented. If implemented in the future, SONGS will institute a surveillance program where, at 5-year intervals, 1 percent of the GT-Inserts will be inspected for any material degradation. The allowance for GT-Inserts and the commitment to the associated inspection program are described in Section 2.3.3.1.2.4.2 of the SE for Amendment Nos. 213 and 205 for SONGS Units 2 and 3, respectively (ADAMS Accession No. ML072550175).

The licensee stated that its current plans are to have all the spent fuel currently stored in the SFPs transferred to the dry cask storage ISFSI before the operating license for either SONGS Units 2 or 3 expires. However, SCE stated it will develop a list of long-lived, passive structures and components if unforeseen circumstances threaten to extend the period of fuel storage in the SFP beyond the current licensed period. SCE will develop the list and an associated aging-management program for those components if all of the spent fuel has not been removed from the SFP by February 16, 2021.

The expiration date of the Unit 2 operating license (that is, the end of the initial 40-year period of operation) is February 16, 2022. The expiration date of the Unit 3 operating license is November 15, 2022. All spent fuel onsite is expected to be moved to the ISFSI approximately 3 years prior to the expiration of the initial 40-year period of operation for both Units 2 and 3. Therefore, for the Units 2 and 3 SFPs, there is no anticipated need for long-lived, passive structures and components beyond the 40-year period of operation for Units 2 and 3, nor is there an anticipated need to monitor or maintain such structures and components beyond the licensed 40-year period of operation. Should the transition of fuel to the ISFSI be delayed by unforeseen events, it is possible that spent fuel could remain in the SFPs beyond the expiration of the 40-year operating period. Therefore, SCE proposed new license conditions for SONGS Units 2 and 3.

New License Condition 2.C.(28) for SONGS Unit 2, and License Condition 2.C.(29) for SONGS Unit 3, will read:

<u>Unit 2</u>

(28) Prior to February 16, 2021, if all spent fuel has not been removed from the Unit 2 spent fuel pool, an aging-management program shall be submitted for NRC approval. The scope of the program shall include those long-lived, passive structures and components that are needed to provide reasonable assurance of the safe condition of the spent fuel in the spent fuel pool. Once approved, the program shall be described in the Updated Final Safety Analysis Report and shall remain in effect for Unit 2 until such time that all spent fuel has been removed from the Unit 2 spent fuel pool.

<u>Unit 3</u>

(29) Prior to February 16, 2021, if all spent fuel has not been removed from the Unit 3 spent fuel pool, an aging-management program shall be submitted for NRC approval. The scope of the program shall include those long-lived, passive structures and components that are needed to provide reasonable assurance of the safe condition of the spent fuel in the spent fuel pool. Once approved, the program shall be described in the Updated Final Safety Analysis Report and shall remain in effect for Unit 3 until such time that all spent fuel has been removed from the Unit 3 spent fuel pool.

The NRC staff has evaluated the licensee's proposed response to the maintenance of long-lived passive structures and components considering the following applicable NRC regulations:

The regulation in 10 CFR Part 50, Appendix A, General Design Criteria (GDC) 62, "Prevention of criticality in fuel storage and handling," requires the prevention of criticality by physical systems or processes, preferably by use of geometrically safe configurations.

The regulations in 10 CFR 50.51(b) require licensees that have provided certifications for permanent cessation of power operations and permanent removal of fuel in accordance with 10 CFR 50.82(a)(1)(i) and 10 CFR 50.82(a)(1)(ii) to take actions necessary to decommission and decontaminate the facility and continue to maintain the facility in a safe condition.

The regulations in 10 CFR 50.65, "Requirements for monitoring the effectiveness of maintenance at nuclear power plants," require licensees to monitor performance or condition of SSCs to ensure they are capable of fulfilling their intended function. The scope of the monitoring specified in 10 CFR 50.65(a)(1) applies to safety-related SSCs as stated in 10 CFR 50.65(b)(1) and to nonsafety-related SSCs whose failure could prevent safety-related SSCs from fulfilling their intended function as stated in 10 CFR 50.65(b)(2)(ii).

The regulations in 10 CFR 50.68 specify requirements for the prevention of criticality accidents and mitigating the radiological consequences of a criticality accident.

The licensee has proposed aging management related license conditions for both SONGS Units 2 and 3, contingent that all remaining fuel will be removed from the SFP by February 16, 2021. If by this time the fuel is not removed from the SFP, the license condition will require that the licensee submit an aging management program for NRC approval. The scope of the program shall include those long lived, passive structures and components that are needed to provide reasonable assurance of the safe condition of the spent fuel in the SFP. Once approved, the program shall be described in the UFSAR and shall remain in effect until such time that all spent fuel has been removed from the SFP. The NRC staff notes that the proposed changes do not affect the design or use of the existing fuel racks, and therefore no criticality analysis was made in association with the changes. The proposed changes also keep intact the systems for the SFP needed to keep the fuel in a subcritical condition. The staff has reviewed the licensee's response to the staff's aging-management concerns and the proposed license conditions to address the concerns. Given that the licensee expects to have all fuel removed from the SONGS SFPs prior to the expiration of the original operating license, the NRC staff has concluded that the proposed new License Condition 2.C.(28) for SONGS Unit 2 and License Condition 2.C.(29) for SONGS Unit 3, adequately address the staff's concerns regarding the maintenance of passive, long-lived structures and components in a safe condition beyond the normal licensed operating period of 40 years, and therefore, finds that the new license conditions are acceptable.

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3.8.8 Deletion of License Condition 2.J and Proposed New License Condition 3

Current License Condition 2.J, reads:

Unit 2

J. This license is effective as of the date of issuance and shall expire at midnight on February 16, 2022.

Unit 3

J. This license is effective as of the date of issuance and shall expire at midnight on November 15, 2022.

Revised License Condition 2.J would state for SONGS Units 2 and 3, will read:

J. Deleted

SCE stated that this license condition can be deleted because SONGS Units 2 and 3 have permanently ceased operation. 10 CFR 50.82(a)(2) prohibits operation of the SONGS Units 2 and 3 reactor since the certifications described therein have been docketed. SCE has proposed that this license condition be replaced by new License Condition 3, which conforms to 10 CFR 50.51, "Continuation of license," in that the license authorizes ownership and possession of SONGS Units 2 and 3 until the Commission notifies the licensee in writing that the license is terminated. The proposed new license condition for SONGS Units 2 and 3, to be used in place of License Condition 2.J., will be License Condition 3.

New License Condition 3 for SONGS Unit 2, will read:

3 On June 12, 2013, Southern California Edison (SCE) certified that operations at San Onofre Nuclear Generating Station Unit 2 would permanently cease in accordance with 10 CFR 50.82(a)(1)(i). On July 22, 2013, SCE certified that the fuel had been permanently removed from the reactor vessel in accordance with 10 CFR 50.82(a)(1)(ii). As a result, the 10 CFR 50 license no longer authorizes operation of the reactor, or the emplacement or retention of fuel in the reactor vessel.

This license is effective as of the date of issuance and authorizes ownership and possession of San Onofre Nuclear Generating Station Unit 2 until the Commission notifies the licensee in writing that the license is terminated. The licensee shall:

A. Take actions necessary to decommission the plant and continue to maintain the facility, including, where applicable, the storage, control and maintenance of the spent fuel, in a safe condition; and

B. Conduct activities in accordance with all other restrictions applicable to the facility in accordance with the NRC regulations and the applicable provisions of the 10 CFR 50 facility license as defined in Section 2 of this license.

New License Condition 3 for SONGS Units 3, will read:

3. On June 12, 2013, Southern California Edison (SCE) certified that operations at San Onofre Nuclear Generating Station Unit 3 would permanently cease in accordance with 10 CFR 50.82(a)(1)(i). On June 28, 2013, SCE certified that the fuel had been permanently removed from the reactor vessel in accordance with 10 CFR 50.82(a)(1)(ii). As a result, the 10 CFR 50 license no longer authorizes operation of the reactor, or the emplacement or retention of fuel in the reactor vessel.

This license is effective as of the date of issuance and authorizes ownership and possession of San Onofre Nuclear Generating Station Unit 3 until the Commission notifies the licensee in writing that the license is terminated. The licensee shall:

A. Take actions necessary to decommission the plant and continue to maintain the facility, including, where applicable, the storage, control and maintenance of the spent fuel, in a safe condition; and

B. Conduct activities in accordance with all other restrictions applicable to the facility in accordance with the NRC regulations and the applicable provisions of the 10 CFR 50 facility license as defined in Section 2 of this license.

The NRC staff has reviewed the proposed deletion of Licensee Condition 2.J and the proposed new License Condition 3 and determined that License Condition 2.J, which documented the date of the expiration of the license, is no longer meaningful for the permanently shutdown condition of the plant in the process of decommissioning. The proposed new License Condition 3 documents the current condition of the plant and summarizes the actions and requirements applicable to the facility by regulation. The proposed License Condition 3 is consistent with the regulatory requirements applicable to the facility in the permanently shutdown and defueled condition, and consistent with a previously issued license conditions for the permanently shutdown and defueled Millstone Unit 1 and the Kewaunee Power Station. Based on the above, the NRC staff finds that the proposed license condition changes are acceptable.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the California State official was notified on May 28, 2015, of the proposed issuance of the amendments. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendments change a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes surveillance requirements. The NRC staff has determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration, and there has been no public comment on such finding as published in the *Federal Register* on September 16, 2014 (79 FR 55513). The amendments also relates to changes in recordkeeping, reporting, or administrative procedures or requirements. Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9) and 10 CFR 51.22(c)(10). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

6.0 CONCLUSION

The NRC staff has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) there is reasonable assurance that such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

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Date: July 17, 2015

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A copy of the Safety Evaluation is also enclosed. The Notice of Issuance will be included in the Commission's biweekly *Federal Register* notice.

Sincerely,

/RA/

Thomas J. Wengert, Senior Project Manager
Plant Licensing IV-2 and Decommissioning
Transition Branch
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-361 and 50-362

Enclosures:
1. Amendment No. 230 to NPF-10
2. Amendment No. 223 to NPF-15
3. Safety Evaluation

cc w/enclosures: Distribution via Listserv

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DNold, NRR

ADAMS Accession No.: ML15139A390 * see previous **concurrence via memo

OFFICE	NRR/DORL/LPL4-2/PM	NRR/DORL/LPL4-2/LA	NRR/DRA/ARCB/BC**	NRR/DSS/STSB/BC**	NRR/DSS/SCVB/BC**
NAME	WHuffman*	PBlechman*	UShoop (by JDozier for)	RElliott* (by MChernoff for)	RDennig
DATE	5/21/15	7/13/15	02/13/15	4/2/15	9/30/14
OFFICE	NRR/DSS/SBPB/BC**	NRR/DSS/SRXB/BC**	NRR/DE/EEEB/BC**	NRR/DE/EICB/BC**	OGC/NLO
NAME	GCasto	CJackson	JZimmerman	JThorp	BMizuno*
DATE	3/16/15	3/27/15	6/1/15	2/27/15	7/14/15
OFFICE	NRR/DORL/LPL4-2/BC	NRR/DORL/LPL4-2/PM			
NAME	MKhanna	TWengert			
DATE	7/17/15	7/17/15			

OFFICIAL RECORD COPY

CERTIFICATE OF SERVICE

I hereby certify that on July 20, 2020, I electronically filed the foregoing with the Clerk of the Court for the United States Court of Appeals for the Ninth Circuit by using the appellate CM/ECF system.

Participants in the case who are registered CM/ECF users will be served by the appellate CM/ECF system.

Dated: July 20, 2020

Respectfully submitted,
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No. 20-70899

**IN THE UNITED STATES COURT OF APPEALS
FOR THE NINTH CIRCUIT**

IN RE PUBLIC WATCHDOGS,

Petitioner,

v.

UNITED STATES NUCLEAR REGULATORY COMMISSION,

Respondent.

SOUTHERN CALIFORNIA EDISON COMPANY,
Intervenor.

INTERVENOR SOUTHERN CALIFORNIA EDISON'S SUPPLEMENTAL EXCERPTS OF RECORD

VOLUME 2 OF 8

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SUPPLEMENTAL EXCERPTS OF RECORD:

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Includes Errata dated June 5, 1981.

NUREG-0490

Final Environmental Statement

related to the operation of San Onofre Nuclear Generating Station, Units 2 and 3

Docket Nos. 50-361 and 50-362

Southern California Edison Company
San Diego Gas & Electric Company
The City of Riverside
The City of Anaheim

U.S. Nuclear Regulatory Commission

Office of Nuclear Reactor Regulation

April 1981

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U.S. Nuclear Regulatory Commission
Washington, DC 20555

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

JUN 5 1981

Docket Nos.: 50-361/362

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Gentlemen:

SUBJECT: ISSUANCE OF ERRATA TO FINAL ENVIRONMENTAL STATEMENT (SAN ONOFRE NUCLEAR GENERATING STATION, UNITS 2 AND 3)

The Nuclear Regulatory Commission has issued the enclosed errata to the Final Environmental Statement (FES) related to the San Onofre Nuclear Generating Station, Units 2 and 3. Although implicit in the FES, this errata clarifies the staff's consideration of reasonable alternatives to the proposed action. The enclosed discussion should be added to Section 10.1.

Sincerely,

Frank J. Miraglia

Frank J. Miraglia, Acting Chief
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Enclosure:
Errata (20 copies)

cc: See next page.

ERRATA

FINAL ENVIRONMENTAL STATEMENT

SAN ONOFRE NUCLEAR GENERATING STATION, UNITS 2 AND 3

ALTERNATIVES TO THE PROPOSED ACTION

During the construction permit stage, the staff analyzed alternative sites, plant designs, and methods of power generation, including the alternative of not adding the production capacity. The staff concluded, based on its analysis of these alternatives, as well as on a cost-benefit basis, that additional capacity was needed that a nuclear-fueled plant would be environmentally acceptable, and that SONGS, at a specified site and of a specified design, were acceptable from both economic and environmental perspectives. Since that time, construction of SONGS has been nearly completed and many of the economic and environmental costs associated with the construction of the facility have already been incurred and must be viewed as "sunk costs" in any prospective assessment.

The staff believes that the only reasonable alternative to the proposed action of issuance of operating licenses for SONGS appropriately considered at this stage is denial of the operating licenses for the facility, thereby not permitting the addition of the essentially built generating capacity to the applicant's generating system. Alternatives such as construction of the units at another site, extensive modifications to the facility, or construction of facilities utilizing different energy sources would each require additional construction activity with its accompanying economic and environmental costs. Therefore, unless major safety or environmental concerns resulting from operation of SONGS are revealed that were not evident and considered during the construction permit review, these alternatives are unreasonable as compared to operating the already constructed facility. No such concerns have been identified with respect to operation of SONGS.

The continued need for the capacity to be generated by SONGS is discussed in section 8 of this FES.

Accordingly, the staff concludes that the preferrable alternative is operation of SONGS.

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NUREG-0490

Final Environmental Statement

related to the operation of San Onofre Nuclear Generating Station, Units 2 and 3

Docket Nos. 50-361 and 50-362

Southern California Edison Company
San Diego Gas & Electric Company
The City of Riverside
The City of Anaheim

U.S. Nuclear Regulatory Commission

Office of Nuclear Reactor Regulation

April 1981

SUMMARY AND CONCLUSIONS

This Environmental Statement was prepared by the U.S. Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation (hereinafter referred to as the staff).

1. The action is administrative.

2. The proposed action is the issuance of Operating Licenses jointly to the Southern California Edison Company (SCE) and the San Diego Gas and Electric Company (SDG&E) for the startup and operation of Units 2 and 3 of the San Onofre Nuclear Generating Station, adjacent to San Onofre Unit 1, located on the Pacific coast in the State of California, County of San Diego (Docket Nos. 50-361 and 50-362).

   The City of Anaheim, California, and the City of Riverside, California, have recently been added as co-holders of the Construction Permits for San Onofre 2 and 3, and will soon request to be included as applicants for Operating Licenses. The four groups are co-owners of the facility, and are referred to herein as the applicant.

   Both units will employ pressurized water reactors to produce up to 3410 thermal megawatts (MWt) each. Steam turbine-generators will use this heat to provide a net power output of up to 1106 electrical megawatts (MWe) each. The exhaust steam will be cooled by once-through flow of water pumped from the Pacific Ocean and returned to it through a diffuser-type system.

3. The information in this statement represents the second assessment by the staff of the environmental impacts associated with the San Onofre Nuclear Generating Station, Unit Nos. 2 and 3, pursuant to the requirements of the National Environmental Policy Act (NEPA) of 1969 and 10 CFR Part 51 of the Commission's Regulations. After receipt of an application (1970) to construct this plant, the staff carried out a review of impacts that would occur during the construction and operation of this plant. This evaluation was issued as a Final Environmental Statement in March 1973. As a result of this environmental review, a staff safety review, an evaluation by the Advisory Committee on Reactor Safeguards, and a public hearing in San Diego, California during January 16-24, 1973 and May 14-22, 1973, and in San Clemente, California, during March 13-15, 1973, the U.S. Atomic Energy Commission (AEC) [now Nuclear Regulatory Commission (NRC)] issued permits in October 1973 for the construction of Units 2 and 3. As of December 1980, Unit 2 was approximately 97% complete and Unit 3 was approximately 68% complete. The applicant has applied for licenses to operate the nuclear units and has submitted the required safety and environmental reports to support this application (March 1977). The staff has reviewed the activities associated with the proposed operation of these units and their potential impacts, both beneficial and adverse, are summarized as follows:

   a. Cooling water heated to about 11°C (20°F) above inlet temperature will be discharged from each unit to the Pacific Ocean at a rate of about 53 m^3/s (846,000 gpm) (Sect. 3.2.2). The heated water may result in the destruction of at least a portion of the San Onofre Kelp Bed during the summer months. However, the long-term thermal impacts are not likely to be severe (Sect. 5.4.2.1) and violations of the state thermal standards are unlikely (Sect. 5.3.1).

   b. An impact on aquatic resources may occur in the cooling water intake structure through entrainment of plankton and impingement of fish. These losses are not expected to have a significant impact on the overall biotic populations in the area.

   c. Chemical effluents from Units 2 and 3 should cause only minimal impact in the area of the discharge, and no significant impact on the aquatic biota in the Pacific Ocean (Sect. 5.4.2.2).

   d. The program for operation and maintenance of transmission lines has been designed to reduce environmental impact. Existing transmission lines and towers will be used where possible. About 7.2 ha (17.8 acres) will be occupied by new towers, access roads, and switchyards (Sect. 2.2.2).

   e. About 16 ha (40 acres) of coastal land which could otherwise have been used primarily for recreation or maintained as wildlife habitat will be occupied by Units 2 and 3 (Sect. 2.2.2).

f. The removal of approximately 1.4 km (0.85 mile) of beach from unrestricted public use, as required by the Construction Permit, is a significant cost of operation.

g. No detectable impacts are anticipated from releases of radioactive materials as a consequence of normal operation (Sect. 5.5.1.6).

h. The risk associated with accidental radiation exposure is very low (Sect. 7).

i. Nothing of known local historic or archaeological interest will be disturbed on the plant site by the operation of Units 2 and 3. A survey along the transmission right-of-way evaluated 41 archaeological sites; of these 23 will be nominated for inclusion in the National Register of Historic Places (Sect. 5.2).

4. The following Federal and State agencies were asked to comment on the Draft Environmental Statement:

- Department of Agriculture
- Department of the Army (Corps of Engineers)
- Department of Commerce
- Department of Energy
- Department of the Interior
- Department of Health, Education and Welfare
- Department of Housing and Urban Development
- Department of Transportation
- Environmental Protection Agency
- Federal Energy Regulatory Commission
- Advisory Council on Historic Preservation
- California Department of Health (Water Pollution Control Commission, Air Pollution Control Commission, Occupational Health Office)
- California Department of Natural Resources
- California Department of Parks and Recreation

Comments on the Draft Environmental Statement were received from the following:

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- Department of Agriculture, Science and Education Administration
- Department of Agriculture, Soil Conservation Service
- Department of the Army, Corps of Engineers
- Department of Commerce
- Department of Energy, Federal Energy Regulatory Commission
- Department of Health, Education and Welfare
- Department of Housing and Urban Development
- Department of the Interior
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- Richard J. Wharton
- Union of Concerned Scientists
- Southern California Edison Company
- Frank H. Grundel
- San Diego Association of Governments

Copies of these comments are appended to this Final Environmental Statement as Appendix A. The staff has considered these comments, and the responses are located in Section 11.

5. This Final Environmental Statement was made available to the public, to the Environmental Protection Agency, and to other specified agencies in April 1981.

6. On the basis of the analysis and evaluation set forth in this statement, and after weighing the environmental, economic, technical and other benefits against environmental costs and after considering available alternatives at the construction stage, it is concluded that the action called for under NEPA and 10 CFR Part 51 is the issuance of operating licenses for Units 2 and 3 of the San Onofre Nuclear Generating Station subject to the following conditions for the protection of the environment:

   (A) License Conditions
   Before engaging in activities that may result in a significant adverse environmental impact that was not evaluated or that is significantly greater than evaluated in this Environmental Statement, the licensee shall provide written notification of such activities to the Office of Nuclear Reactor Regulation and receive written approval from that office before proceeding with such activities.

   (B) Significant Environmental Technical Specification Requirements

      (1) If, during the operating life of the Station, effects or evidence of potential irreversible damage are detected, the licensee will provide to the staff an analysis of the problem and a proposed course of action to alleviate the problem.

      (2) The licensee will carry out the operational environmental monitoring programs outlined in Section 6.

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FOREWORD

This environmental statement was prepared by the U.S. Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation (hereinafter referred to as the staff) in accordance with the Commission's regulations, 10 CFR 51, which implement the requirements of the National Environmental Policy Act of 1969 (NEPA).

The NEPA states, among other things, that it is the continuing responsibility of the Federal government to use all practicable means, consistent with other essential considerations of national policy, to improve and coordinate Federal plans, functions, programs, and resources to the end that the Nation may:

- Fulfill the responsibilities of each generation as trustee of the environment for succeeding generations.
- Assure for all Americans safe, healthful, productive, and aesthetically and culturally pleasing surroundings.
- Attain the widest range of beneficial uses of the environment without degradation, risk to health or safety, or other undesirable and unintended consequences.
- Preserve important historic, cultural, and natural aspects of our national heritage, and maintain, wherever possible, an environment that supports diversity and variety of individual choice.
- Achieve a balance between population and resource use which will permit high standards of living and a wide sharing of life's amenities.
- Enhance the quality of renewable resources and approach the maximum attainable recycling of depletable resources.

Further, with respect to major Federal actions significantly affecting the quality of the human environment, Sect. 102(2)(C) of the NEPA calls for preparation of a detailed statement on:

(i) the environmental impact of the proposed action;

(ii) any adverse environmental effects which cannot be avoided should the proposal be implemented;

(iii) alternatives to the proposed action;

(iv) the relationship between local short-term uses of man's environment and the maintenance and enhancement of long-term productivity; and,

(v) any irreversible and irretrievable commitments of resources which would be involved in the proposed action should it be implemented.

An environmental report accompanies each application for a construction permit or for a full-power operating license. A public announcement of the availability of the report is made. Any comments by interested persons on the report are considered by the staff. In conducting the required NEPA review, the staff meets with the applicant to discuss items of information in the environmental report, to seek new information from the applicant that might be needed for an adequate assessment, and generally to ensure that the staff has a thorough understanding of the proposed project. In addition, the staff seeks information from other sources that will assist in the evaluation and visits and inspects the project site and surrounding vicinity. Members of the staff may meet with state and local officials who are charged with protecting state and local interests. On the basis of all the foregoing and other such activities or inquiries as are deemed useful and appropriate, the staff makes an independent assessment of the considerations specified in Sect. 102(2)(C) of the NEPA and 10 CFR Part 51.

This evaluation leads to the publication of a draft environmental statement, prepared by the Office of Nuclear Reactor Regulation, which is then circulated to Federal, state, and local governmental agencies for comment. A summary notice of the availability of the applicant's environmental report and the draft environmental statement is published in the Federal Register. Interested persons are also invited to comment on the proposed action and on the draft statement.

After receipt and consideration of comments on the draft statement, the staff prepares a final environmental statement, which includes a discussion of questions and concerns raised by the comments and the disposition thereof; a final benefit-cost analysis, which considers and balances the environmental effects of the facility and the alternatives available for reducing or avoiding adverse environmental effects with the environmental, economic, technical, and other benefits of the facility; and a conclusion as to whether – after the environmental, economic, technical, and other benefits are weighed against environmental costs and after available alternatives have been considered – the action called for, with respect to environmental issues, is the issuance or denial of the proposed permit or license or its appropriate conditioning to protect environmental values. This final environmental statement and the safety evaluation report prepared by the staff are submitted to the Atomic Safety and Licensing Board for its consideration in reaching a decision on matters in controversy regarding the application. The same format as used in the Draft Environmental Statement is used in this Final Statement to facilitate its review.

This environmental review deals with the impact of operation of San Onofre Nuclear Generating Station Units 2 and 3 (SONGS 2 & 3). Assessments that are found in this statement supplement or modify those described in the Final Environmental Statement (FES-CP) that was issued in March 1973 in support of issuance of construction permits for the units. The information found in the various sections of this Statement updates the FES-CP in four ways: (1) by identifying differences between environmental effects of operation (including those which would enhance as well as degrade the environment) currently projected and the impacts that were described in the preconstruction review, (2) by reporting the results of studies that had not been completed at the time of issuance of the FES-CP and that were required by the NRC staff to be completed before initiation of the operational review, (3) by evaluating the applicant's preoperational monitoring program and by factoring the results of this program into the design of a postoperational surveillance program and into the development of environmental technical specifications, and (4) by identifying studies being performed by the applicant that will yield additional information relevant to the environmental impacts of operating SONGS 2 & 3.

Copies of this statement are available for inspection at the Commission's Public Document Room, 1717 H Street, N.W., Washington, D.C.; the Mission Viejo Branch Library, 24851 Chrisanta Drive, Mission Viejo, California; and the NRC Office of Inspection and Enforcement, 1990 N. California Boulevard, Walnut Creek, California. Copies of this statement may be obtained as indicated on the inside front cover. Mr. Dino C. Scaletti is the NRC Project Manager for this statement. Mr. Scaletti may be contacted at (301) 492-8443.

1. INTRODUCTION

1.1 HISTORY

On May 28, 1970, the Southern California Edison Company and the San Diego Gas and Electric Company filed an application with the Atomic Energy Commission (now Nuclear Regulatory Commission) for permits to construct San Onofre Nuclear Generating Station Units 2 and 3 (SONGS 2 & 3). Construction Permits Nos. CPPR-97 (Unit 2) and CPPR-98 (Unit 3) were issued on October 18, 1973, following reviews by the AEC regulatory staff and the Commission's Advisory Committee on Reactor Safeguards, as well as a public hearing before an Atomic Safety and Licensing Board in San Diego and San Clemente, California on January 16 to 24, March 13 to 15, and May 14 to 22, 1973. An additional session of the hearing was held in Los Angeles, California on May 19, 20, and 21, 1976. The conclusions reached in the staff's environmental review were issued in a Final Environmental Statement (FES-CP) in March 1973.

As of December 1980, construction of Unit 2 was about 97% complete and construction of Unit 3 was about 68% complete. Each unit has a pressurized-water reactor that will produce up to 3410 MWt and a net electrical output of up to 1106 MWe.

In November 1976 Southern California Edison Company and San Diego Gas and Electric Company submitted an application including a Final Safety Analysis Report (FSAR) and Environmental Report (ER) requesting issuance of operating licenses for Units 2 and 3. These documents were docketed on March 22, 1977, and the operational safety and environmental reviews were initiated at that time.

The City of Anaheim, California, and the City of Riverside, California have recently been added as co-holders of the Construction Permits for San Onofre 2 and 3 and will soon request to be included as applicants for Operating Licenses. The four groups are co-owners of the facility and are referred to herein as the applicant.

1.2 PERMITS AND LICENSES

The applicant has provided a status listing of environmentally related permits, approvals, licenses, etc., which are required from Federal, regional, state, and local agencies in connection with the proposed project (ER, Sect. 12). The staff has reviewed that listing. An amendment to the permit from the California Coastal Commission may be required to obtain approval for the modified exclusion area plan. The staff is not aware of any other potential non-NRC licensing difficulties that would significantly delay or preclude the proposed operation of the plant.

2. THE SITE

2.1 RESUME

The staff visited the SONGS site in May 1977 primarily to determine what changes had occurred at the site and in surrounding areas since the preconstruction environmental review in late 1972. In addition, more detailed information about the operation of SONGS 2 & 3 was obtained as a result of this visit.

Population distribution estimates have been updated and extended to the year 2020. The major land use change has been the construction of the plant itself. Transmission line routes have undergone some changes.

An updated description of the surface-water hydrology is given in Sect. 2.3.1.

The section on meteorology has been revised to include the results of recent observations.

Considerable additional field work and sampling is reflected in the description of terrestrial and aquatic ecology in Sect. 2.5.

2.2 REGIONAL DEMOGRAPHY AND LAND USE

2.2.1 Population change

Population for 1976 by sectors within 80 km (50 miles) of the plant and the projected population estimates to the year 2020 are provided in Tables 2.1-2 through 2.1-15 of the ER. The population within a 16-km (10-mile) radius of the site in 1976 was 57,241. By 1980 this population was expected to increase to 67,547 - an annual growth rate of 4.2% (ER, Sect. 2.1.3.2.1). The major cities in the area and their 1975 populations are San Clemente (20,794), 6.4 km (4 miles) northeast; San Juan Capistrano (13,658), 16.8 km (10.5 miles) northwest; Oceanside (54,900), 27.2 km (17 miles) southeast; and San Diego (1,518,000), 81.6 km (51 miles) southeast. Table 2.1 provides 1976 population data by sector within 16 km (10 miles) of the site.

Table 2.1. Population by sector and distance with 10 miles of San Onofre site (1976)

Sector	Distance (miles)						
	0 to 1	1 to 2	2 to 3	3 to 4	4 to 5	5 to 10	Total 0 to 10
W	0	0	0	0	0	0	0
WNW	0	0	0	0	0	0	0
NW	0	656	54	3532	5298	21,979	31,519
NNW	0	732	630	0	0	6,541	7,903
N	0	0	0	4300	0	519	4,819
NNE	0	0	0	0	0	0	0
NE	0	0	4600	0	0	0	4,600
ENE	0	0	0	0	0	0	0
E	0	0	0	0	4300		4,300
ESE	0	0	0	0	0	3,100	3,100
SE	0	0	0	0	0	1,000	1,000
SSE	0	0	0	0	0	0	0
Total	0	1388	5284	7832	9598	33,139	57,241

Source: ER, Table 2.1-2.

(To convert miles to kilometers, multiply by 1.6.)

Table 2.2 presents projected population and annual growth rates within 16 km (10 miles) of the plant between 1976 and 2020. The total percentage change in population for the area between 1976 and 2020 is projected to be 99.4%. These projections are based on surveys made by the Southern California Association of Governments, the Comprehensive Planning Organization of San Diego County, the California State Department of Finance, and the applicant (ER, Sect. 2.1.3.2.3).

Table 2.2. Projected population and annual growth rate within 16 km of the San Onofre site

Year	Projected population	Annual growth[a] (%)		Change (%)
1976 1980	57,241 67,547	4.2	1.6	99.4
1980 1990	67,547 89,521	2.9		
1990 2000	89,521 91,949	0.3		
2000 2010	91,949 101,945	1.0		
2010 2020	101,945 114,139	1.1		

[a]Compounded annually.

Source: Adapted from ER, Table 2.1-8.

2.2.2 Changes in land use

Since issuance of the FES-CP in 1973, the construction of SONGS 2 & 3 is the only major change in land use in the site vicinity. Site preparation required the excavation of 16.39 ha (40.5 acres) of the San Onofre Bluffs, which otherwise could be used primarily for recreation. Most of this material was deposited on 34 ha (84 acres) at Japanese Mesa, a relatively flat area just north and across Interstate 5 from the site on Camp Pendleton Marine Base (ER, Sect. 4.1.2). In addition, about 304.8 m (1000 ft) of beach front has remained closed except as a passageway during the construction period (ER, Appendix 12-B, p. 7).

The area within an 8-km (5-mile) radius of the site occupies parts of two counties. The part of this area that lies in Orange County is entirely within San Clemente. The predominant land use in San Clemente is single family residential, light commercial, and recreational. Industrial land use in San Clemente is limited to light industry only. Because the available developable land is steep, future development in that area is expected to be slow with only low residential densities permitted by the city (ER, Sect. 2.1.4.3.1). In San Diego County, the 8-km (5-mile) radius area lies within Camp Pendleton Marine Base. About 95% of Camp Pendleton is unimproved land that is used for military purposes, recreation, and conservation (FES-CP, Sect. 2.2.2). Figure 2.1-12 of the ER provides a detailed land use map of the area within an 8-km (5-mile) radius of the site.

Heavy-haul components for the plant arrive by barge or by vessel at the Del Mar Boat Basin near Oceanside, about 22.5 km (14 miles) south of the site (ER, Suppl. 2, Item 37). The haul route, which was not available at the time the FES-CP was issued, required that a road be cut through the bluffs between the beach and Highway 101, about 11 km (7 miles) north of the Del Mar Boat Basin (ER, Suppl. 2, Item 37).

The description of the transmission lines as presented in Sect. 3.7 of the FES-CP has been modified (Sect. 3.2.5). No new rights-of-way were required: about 5.2 ha (12.8 acres) will be used for new tower bases and for access-road extensions, and 2 ha (5 acres) of land will be covered by the Talega Substation (ER, Suppl. 2, Item 36). Three changes in land use adjacent to the San Onofre–Santiago transmission line route have occurred since the issuance of the FES-CP: (1) construction of a paved road immediately adjacent to a significant portion of the proposed transmission line, (2) bulldozing of a firebreak adjacent to the transmission line on Camp Pendleton Marine Base, and (3) active operation of a large aggregate borrow site adjacent to the line in a third location (ER, Appendix 6A).

2.2.3 Changes in the local economy

Construction activity peaked in late 1979 with an estimated work force of about 3000. The applicant has estimated, after discussions with officials of the labor unions represented at SONGS 2 & 3, that 20%, or about 600 workers, relocated to the southern California area from other parts of the country (ER, p. S.2-167). Although all union craft workers at the site were hired from unions located within a 96-km (60-mile) radius of the site, all of the workers who relocated were travel card members who were assigned by the local unions to SONGS 2 & 3 after the local list was exhausted. Because the construction workers lived throughout the metropolitan areas of San Diego, Orange County, and Los Angeles, the impact of the workers' income was diffuse.

From 1974 through 1976 the applicant estimated that about $4.1 million was spent within a 48-km (30-mile) radius of the site for materials and services. These expenditures accounted for about 0.2% of the total forecast plant cost (ER, p. S.2-174).

2.3 WATER USE

2.3.1 Surface-water hydrology

The only significant water resource in the vicinity of SONGS is the Pacific Ocean. A few streams are located near the site, but these are intermittent.

The currents in the San Onofre vicinity are a superposition of many effects. This current system can be decomposed into individual components. The two most persistent components are the California Current and the tides.

The California Current is evident close to shore and north of Point Conception. However, south of this point the coastline recedes to the east, and water is available for entrainment from the east side of the current. This entrainment tends to make the California Current more diffuse south of Point Conception. Furthermore, the effect of this entrainment in addition to upwelling, winds, and baroclinic instabilities[1] can produce a counter-rotating eddy through the Channel Islands which is known as the Southern California Eddy; the nearshore northward flowing current is the Southern California Countercurrent. Observations indicate that this eddy can exist year-round; however, it is strongest in the fall and in the early winter.

Tides along the California coast are a mixed type with diurnal and semidiurnal components. The diurnal period lasts about 25 hr, and the semidiurnal period is about half the duration of the diurnal. As a result of tidal rotation, flood tide flows up the coast and ebb tide flows down the coast. A more detailed discussion of the tides in the San Onofre vicinity can be found in Sect. 2.6.3 of the FES-CP.

The total near-shore current is the sum of the large-scale current systems, the tides, and other effects such as local winds and offshore storms. The net result is a highly complex current structure that is quite variable in speed and direction. An additional complication is stratification. During the winter when vertical homogeneity exists, near-shore currents are fairly uniform with depth. However, during the summer the presence of the thermocline divides the water column so that only certain components of the net flow are uniform with depth. These components, such as tides, are driven over the entire water column. Surface driving forces (the wind) will penetrate the epilimnion; however, the thermocline represents a barrier to these stresses reaching the hypolimnion. The wind energy is then concentrated in the epilimnion, resulting in an increased intensity of wind-driven flow which can dominate all other components. In contrast, the hypolimnion is relatively free of wind effects and, therefore, is strongly influenced by the tides. The net result is a two-layered flow regime in which the flow in the two layers is only weakly correlated. This already-complicated flow structure can be altered by large amplitude internal waves. The breaking of these waves provides periodic vertical mixing.

A survey of the currents in the San Onofre area was conducted in 1972 by Intersea Research Corporation.[2] Data from this study have been analyzed by Koh and List.[3] From this analysis the following summary information has been extracted.

1. A net drift current can occur in a number of directions; however, the onshore/offshore component of the drift is necessarily smaller than the longshore component.

2. The longshore component of the drift changes direction every 3 to 6 days with downcoast flow typically having a longer duration.

3. The magnitude of the longshore drift is less than 30 cm/sec (0.6 knot).

4. The onshore/offshore component of drift is less than 15 cm/sec (0.3 knot).

5. An upcoast component of drift usually is associated with an onshore component of drift, and vice versa.

6. Both components of tidal flow are typically 10 cm/sec (0.2 knot).

The most detailed study of natural temperature variations in the San Onofre vicinity is that of Koh and List.[3] This study was based on daily temperature measurements from 1966 through 1970 taken at the ends of piers at Balboa, San Clemente, Oceanside, and La Jolla. These data were separated into three frequency ranges - low, middle, and high. The low-frequency component represents data averaged over two months, and it reflects seasonal variations. After removal of these low frequencies, the data were averaged over one week. This is the middle-frequency band, which represents variation within periods from one week to two months. The residual data, the high-frequency band, represents daily to weekly fluctuations. Figure 2.1 is a plot of temperature vs time for the three frequency bands and the raw data for San Clemente. The temperature ranges from 12.1°C (54°F) to 22.9°C (73°F). The low-frequency curve shows an annual temperature cycle with a maximum in midsummer and a minimum in midwinter.

As part of their analysis, Koh and List performed a correlation study among the temperature records from the various locations. Both the low- and middle-frequency ranges showed very high correlations at zero lag time between Oceanside and San Clemente. This indicates that the mechanisms influencing these frequency components have a length scale greater than the distance between the two sampling locations. Therefore, temperature variations at San Onofre within periods of one week or longer can be represented adequately by the corresponding temperature variations at either San Clemente or Oceanside. The correlation of the high-frequency components between these two stations is very weak, indicating that short-term temperature fluctuations are a spatially localized phenomenon. This fact is substantiated by near-surface-temperature measurements made from a moving boat which show that horizontal temperature variations of 1.1°C (2°F) over 1.6 km (1 mile) are not uncommon off the coast of southern California.[3]

An additional feature of the thermal structure in the San Onofre vicinity is vertical stratification. During the winter this region is, in general, isothermal over the water column. As warming progresses, a vertical temperature gradient is established and reaches a maximum in late summer. This natural gradient has been as much as 0.55°C/m (0.3°F/ft).

Ocean salinity in the San Onofre vicinity shows little spatial variation. An annual salinity cycle does exist as a result of annual cycles in the local meteorology and large-scale current systems. During this cycle, salinity typically ranges from 33 to 34 ppt, with the minimum occurring in winter and the maximum occurring in summer.

2.3.2 Groundwater hydrology

The average elevation of the water table at the beach line is +1.5m (+5 ft) mean lower low-water level (MLLW) with a slope of less than 1%; inland, the gradients range from 2 to 8% toward the ocean. Some groundwater can be obtained from the San Onofre Groundwater Basin, and it is used at Camp Pendleton Marine Base, but it is not a resource used by the Station. The Station obtains its domestic supply of freshwater from the Tri-Cities Municipal Water District.

2.3.3 Water quality

Dissolved oxygen concentration in southern California coastal waters ranges from about 5 to 13 mg/liter. Observations at the site vary from 5.4 to 10.0 mg/liter (2 to 3.6 grains/gal). The pH of southern California surface waters varies from 7.5 to 8.4 with a mean of about 8.0.

Measurements of coliform concentrations at the site were made during the period 1967 to 1975. Most of the measurements gave a mean probable number (MPN) of 4 to 43 colonies/100ml (1 to 13 colonies/oz). Only two measurements exceeded 43, and these occurred in 1972 and both gave a MPN value of 460 (140).

Turbidity in the vicinity of the site is due primarily to the suspension of bottom material in the surf zone. Outside the surf zone, turbidity generally decreases as distance from shore increases. Typical depths of Secchi Disc visibility range from 2 to 5 m (6.5 to 16 ft).[4] The vertical variation of turbidity is often quite complex, with alternating layers of clear and turbid water. Visible plumes of turbidity have been observed occasionally on the ocean surface in the vicinity of the Unit 1 offshore discharge structure. These plumes have been observed and, depending on ambient conditions, are caused by the intake and subsequent discharge of naturally turbid water and the entrainment of naturally turbid water into the discharge stream as it moves towards the surface (ER, Sect. 2.4.3.8.2).

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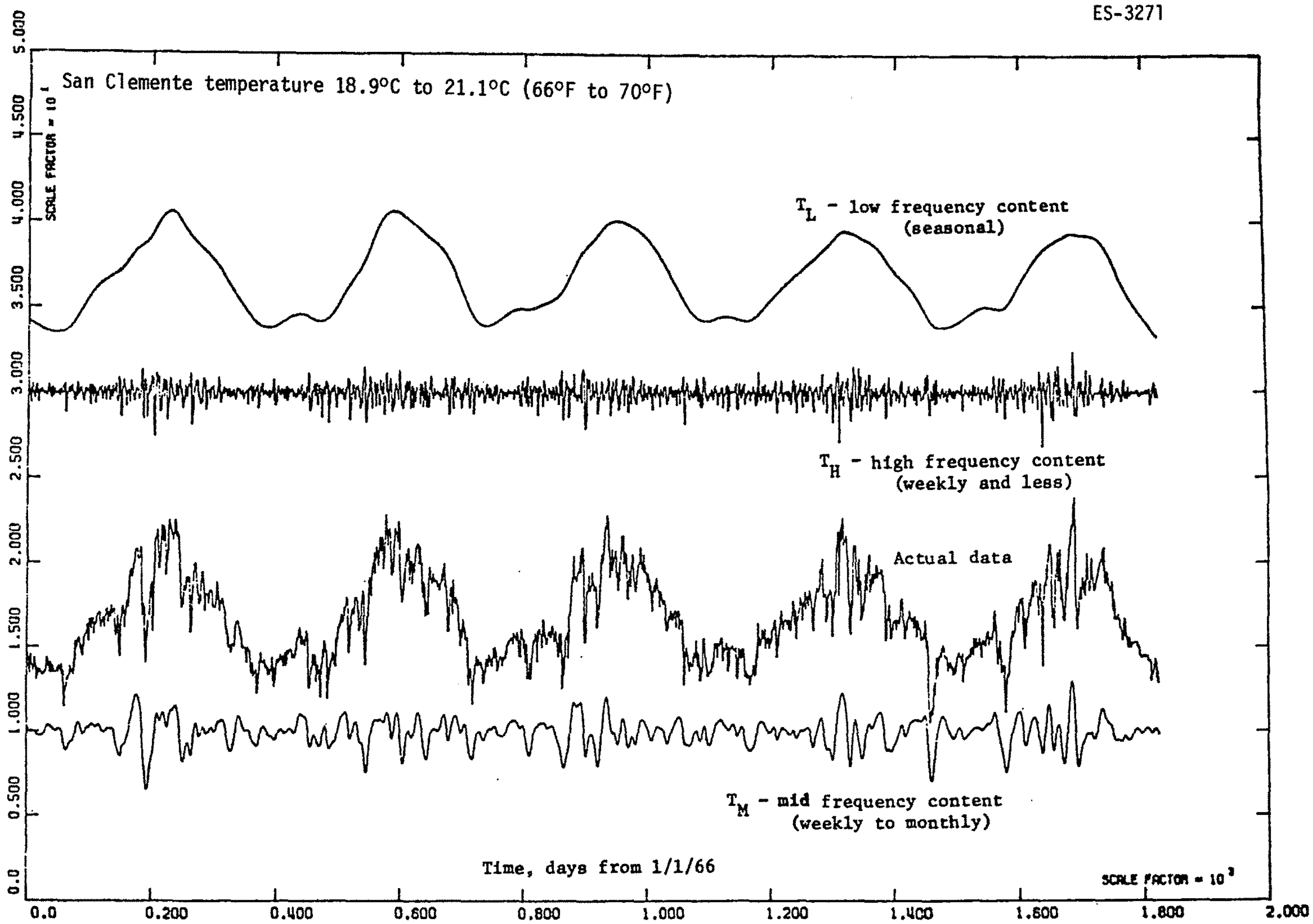

Fig. 2.1. Daily temperatures and decompositions into various frequency bands at San Clemente.
Source: R. C. Y. Koh and E. J. List, *Report to Southern California Edison Company on Further Analysis Related to Thermal Discharges at San Onofre Nuclear Generating Station*, Sept. 30, 1974, Fig. 3.2.

2.3.4 Storm runoff

The probable maximum 1-hr thunderstorm rainfall is 17.8 cm (7.0 in). Much of the country to the north and east of the Station site drains into the San Onofre Creek, which flows into the ocean 2 km (1-1/4 miles) northwest of the site. The land immediately east of the site now drains into a 3.7-m-wide (12-ft-wide) ditch that parallels Interstate Highway 5 (I-5) just east of the Station. Both lanes of I-5 also drain into this ditch, which discharges into San Onofre Creek. Storm runoff from the hills above the site drains through one 182-cm-(72-in.-) and one 107-cm-(42-in.-)diam culvert that run north along the highway right-of-way and then turn under the site to the beach. The culverts and channel are designed for the runoff associated with a 1% chance (100-year) storm. To preclude flooding at the site during the occurrence of a probable maximum thunderstorm, an earthen dike will be constructed to the east side of I-5 to divert runoff and debris from the foothills area to San Onofre Creek.

2.4 METEOROLOGY

2.4.1 Regional climatology[5-9]

The climate of the coastal regions of southern California is strongly influenced by the Pacific Ocean. Summers are relatively cool with daytime temperatures averaging only in the low-to-mid-20s (°C) (70°F); daytime seabreezes are frequent. Outbreaks of hot, dry desert air from east of the coastal mountains (Santa Ana winds) may intrude onto the coastal plain several times each year, primarily in the fall, but temperatures exceed 32°C (90°F) usually less than five days annually. The proximity to the Pacific Ocean also results in mild winters, with daytime highs in the upper teens (°C) (60s°F) and nighttime lows around 5 to 10°C (40s°F). Temperatures below freezing are rare.

Precipitation along the coastal plain averages around 250 mm (10 in.) annually. The rainfall is very seasonally dependent with 85% of the total occurring from November through March; almost no rain falls during the summer months. Average relative humidities range from about 80% during the early morning hours of summer and fall, down to around 55% during winter afternoons.

2.4.2 Local meteorology[5,6,8,9]

The San Onofre site is located on the relatively narrow coastal plain, near the mouth of San Onofre Canyon. Coastal bluffs, nearby hills and valleys, and the Pacific Ocean contribute to the complexity of the site topography. Within 8 km (5 miles) of the site, elevations range from 525 m (1725 ft) above sea level [about 5.5 km (3.5 miles) east of the site] to sea level along the Pacific Ocean.

To assess the local meteorological characteristics of the San Onofre site, climatological data from San Diego, California [80 km (50 miles) southeast of the site]; from Los Angeles, California [95 km (60 miles) northwest]; and data collected onsite are available. These data are reasonably representative of the climatological conditions expected in the vicinity of the site.

In the site area, average daily maximum and minimum temperatures range between 25°C (77°F) and 18°C (64°F) in August, the warmest month, and between 18°C (65°F) and 8°C (46°F) in January, the coolest month. The extreme maximum temperature recorded was 44°C (111°F) at San Diego in September 1963; the extreme minimum temperature was -5°C (23°F) at Los Angeles in January 1937.

The area receives about 250 mm (10 in.) of rain annually; December, January, and February — the wettest three-month period — averages about 150 mm (6 in.), and June, July, and August combined averages less than 2.5 mm (0.1 in.). The maximum 24-hr rainfall recorded among these stations is 157 mm (6.2 in.) at Los Angeles in January 1956. Snowfall is a rarity, with a trace [less than 0.25 mm (0.01 in.)] being the most ever recorded. Heavy fogs [visibility of 0.4 km (0.25 mile) or less] occur on about 30 to 40 days each year along the coast with about half of the occurrences during October through January.

Windflow at the site has a strong diurnal dependence primarily due to the land-sea breeze effect. During daytime hours the windflow has a predominant onshore directional component, whereas at night windflow tends toward a seaward direction. Table 2.3 shows the wind direction with the greatest frequency of occurrence for each hour of the day for the three-year period of January 25, 1973, through January 24, 1976, as measured at the 10-m (33-ft) level of the onsite meteorological tower. Figure 2.2 shows the directional frequency of onsite winds. About 25% of the total windflow over the site was from the northeast and north-northeast (principally night-time offshore flow); 19% of the flow occurred from the west and west-northwest (daytime onshore flow). Winds were calm [windspeeds less than 0.34 m/sec (0.75 mph)] less than 1% of the time at the 10-m (33-ft) level.

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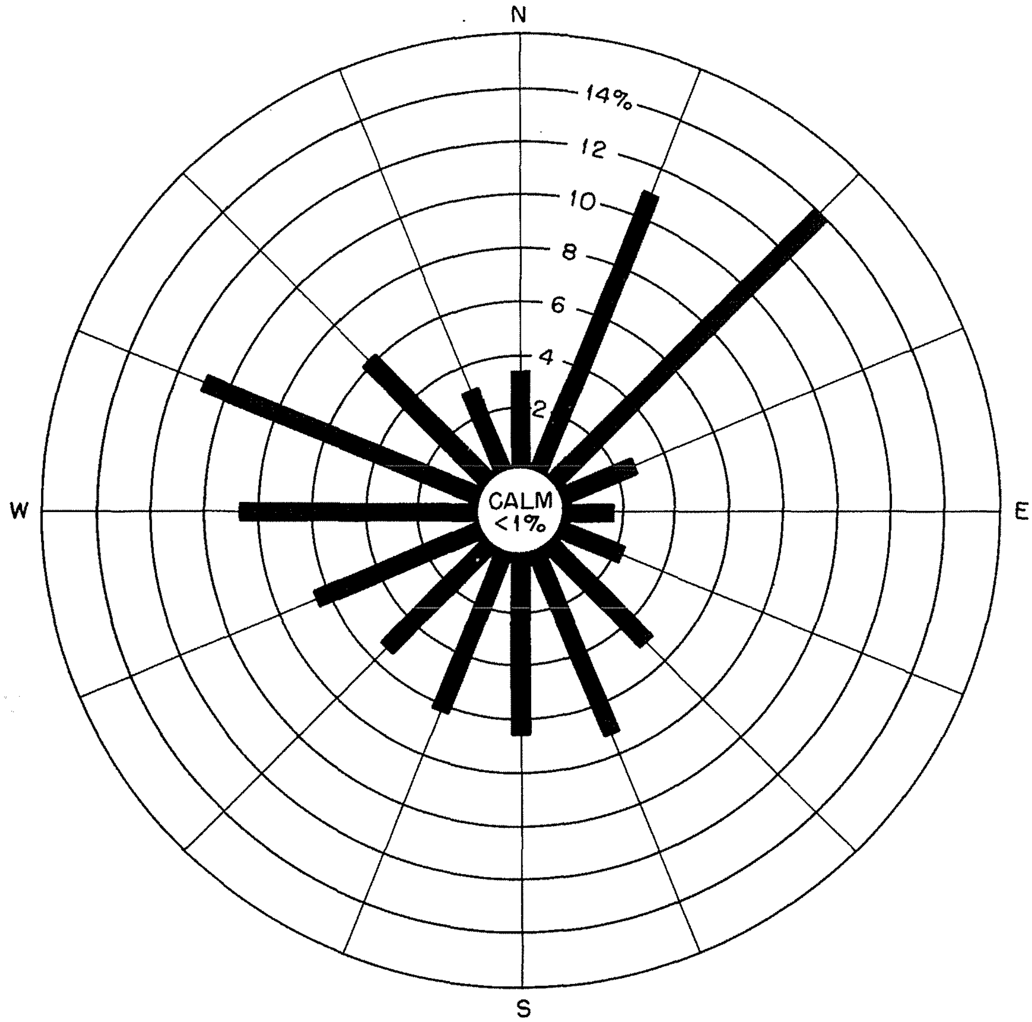

Fig. 2.2. Directional frequency of wind at the San Onofre site. Onsite data at 10 m (33 ft) above ground level, Jan. 25, 1973 through Jan. 24, 1976. Bars show the direction from which the wind blows. Calms are those winds with hourly average speeds less than 0.34 m/sec (0.75 mph).

Table 2.3. Wind direction with greatest frequency of occurrence by time of day at San Onofre Nuclear Generating Station

Data measured at 10-m (33-ft) level of onsite meteorological tower

Hour (AM)	Wind direction	Frequency (%)	Hour (PM)	Wind direction	Frequency (%)
1	NE	28	1	WNW	25
2	NE	26	2	WNW	27
3	NE	27	3	WNW	27
4	NE	28	4	WNW	27
5	NE	30	5	WNW	22
6	NE	30	6	WNW	16
7	NE	25	7	NW	14
8	NE	19	8	NE	13
9	S	12	9	NE	16
10	W	17	10	NE	20
11	W	20	11	NE	23
Noon	WNW	22	Midnight	NE	25

2.4.3 Severe weather[5-13]

Although infrequent, thunderstorms, tornadoes, tropical cyclones, and dust storms can affect the site area. Thunderstorms occur less than 5 days annually. Tropical storms are also rare in the site area, with a storm entering the region less than once every 10 years. The "fastest mile" of wind recorded at Los Angeles was 28 m/sec (62 mph) (March 1952). Snow, glaze, and hail are almost nonexistent in the site vicinity.

Between 1952 and 1975, 23 tornadoes and 21 waterspouts were reported within a 34,000-km^2 (13,000-mi^2) area containing the site. Staff analysis of these tornado data indicates that the mean path area of a tornado in this region is about 0.3 km^2 (0.1 mi^2). Using the methods of Thom, this results in a recurrence interval of 70,000 years for a tornado or waterspout at the plant site.

Dust storms are relatively infrequent within the site region; between 1940 and 1970, dust or blowing dust and sand reduced visibility to under 11 km (7 miles) about 1 hr annually. About 8 days each year there is a high meteorological potential for air pollution.

2.4.4 Atmospheric dispersion[5,6,14,15]

Southern California Edison Company (SCE) has provided joint frequency distributions of wind speed and direction by atmospheric stability class, based on the vertical temperature gradient, collected onsite during the period January 25, 1973 to January 25, 1976. The distributions were for wind speed and direction measured at both the 10- and 40-m (33- and 131-ft) levels with the vertical temperature difference between the 6.1- and 36.6-m (20- and 120-ft) levels. SCE has also conducted a tracer test program to assess the atmospheric dispersion in the landward directions at the San Onofre site. Section 6.2.5 describes the onsite meteorological program and the tracer test program.

The staff has made reasonable estimates of average atmospheric dispersion conditions for SONGS 2 & 3 using an atmospheric dispersion model for long-term releases; this model is based on the "Straight-Line Trajectory Model" described in Regulatory Guide 1.111. The onsite tracer tests showed that ground-level relative concentrations normalized by windspeed were similar whether the source of release was elevated or ground level; thus it was assumed that all plant releases were from ground level. The calculations also include considerations of intermittent releases during more adverse atmospheric dispersion conditions than indicated by an annual average calculation as a function of total duration of release. The calculations include an estimate based on the criteria outlined in Regulatory Guide 1.111 of maximum increase in calculated relative concentration and deposition due to the spatial and temporal variation of the airflow not considered in the straight-line trajectory model. Radioactive decay of effluents and depletion of the effluent plume were also considered as described in Regulatory Guide 1.111.

In the evaluation, we used meteorological data collected onsite between January 25, 1973 and January 24, 1976. All releases were evaluated using joint frequency distributions of wind speed and direction measured at the 10-m (33-ft) level by atmospheric stability [defined by the temperature difference between the 36.6- and 6.1-m (120- and 20-ft) levels]. Data recovery for this time period was 88%.

Table 5.1 presents the calculated values of relative concentration (χ/Q) and relative deposition (D/Q) for specific points of interest.

2.5 SITE ECOLOGY

2.5.1 Terrestrial ecology

The FES-CP describes the terrestrial ecology of the San Onofre site (FES-CP, Sect. 2.8.1). Field work for this description, however, was conducted only during November 1971 and contained very little quantitative data. Consequently, the issuance of the construction permit was subject to the applicant's expansion of its current environmental monitoring program "to determine environmental effects which may occur as a result of site preparation and construction of Units 2 and 3, and to establish an adequate preoperational baseline by which the operational effects of Units 2 and 3 may be judged" (FES-CP, p. iv). In response, the applicant conducted terrestrial ecological studies for a period of 1 year on a 0.61-ha (1.5-acre) quadrat located immediately south of Units 2 and 3 construction site (ER, Appendix 2A). This monitoring program documented seasonal changes in the biotic communities over a 1-year time span and fulfilled the recommendations of NRC Regulatory Guide 4.11.

About 80% of the study area is in a natural plant community of coastal sage scrub, and the remaining 20% has been disturbed by man-related activities. Total cover on the study area ranged from 81 to 98%. The greatest cover was found in February, decreasing toward midsummer. Vegetative diversity in the coastal sage scrub community was relatively low; California sagebrush (*Artemesia californica*) was the dominant species (65% relative cover). Coyote bush (*Baccharis pilularis*) ranked second in the study area (9% relative cover) but had higher relative cover in the disturbed areas than in the climax stand. The applicant's survey suggests that surface disturbances significantly alter the composition of the coastal sage scrub community by encouraging the invasion of exotic perennial and annual plant species, especially mustards and grasses. Establishment of these plants occurred only in areas that have been disturbed (ER, Appendix 2A). As expected for this very small study area (0.61 ha), no endangered plant species were observed.

Fauna observed within the study area included 5 species of reptiles, 12 species of mammals, and 36 species of birds; no amphibians were sighted. None of the species observed in the study area are threatened or endangered as defined by the U.S. Department of the Interior[16] (ER, Sect. 2.2.1.2).

The endangered animal species[16] whose ranges include the vicinity of the plant and associated transmission lines are listed in Table 2.4. Two of these species have been observed by the applicant. The California brown pelican has occurred several times on the beach adjacent to the construction area (ER, Sect. 2.2.1.2), and the California least tern has a nesting colony located near the Del Mar Boat Basin, a facility used by the applicant to move heavy components (see Sect. 2.2.2).

Examination of the geographical distributions[17,18] of the 266 endangered plant species in California[19] indicates that 26 of these species occur in those counties (Orange and/or San Diego) traversed by the transmission lines (Table 2.5). No endangered plant species, however, were observed during the applicant's biological study of the San Onofre-Santiago transmission line route.[20] Biological surveys of the other transmission line routes have not been conducted, but no habitats adjacent to or within the transmission line right-of-way have been classified by state or Federal authorities as being critical to any endangered species (ER, Suppl. 1, Item 22).

2.5.2 Aquatic ecology

The aquatic ecology of the site was described in the FES-CP issued in March 1973, and was based on descriptive data obtained from literature concerning the southern California coast. The FES-CP site description contained minimal baseline information on spatial and temporal differences in species occurrences and population densities. The data obtained since issuance of the FES-CP is primarily from three sources: (1) a thermal effects study performed jointly by Environmental Quality Analysts, Inc., and Marine Biological Consultants, Inc., in 1973 using data and results obtained from 1964-72 by Bendix Marine Advisers, Inc., and Intersea Research Corporation.[21] (2) the SONGS 1 Environmental Technical Specifications (ETS) monitoring program begun in November 1974, conducted by the Lockheed Aircraft Service Company's Department of Marine biology,[22-27]

Table 2.4. Endangered animal species[a] whose ranges include Orange and San Diego counties, California

Common name	Scientific name	Habitat	Reason for decline
California brown pelican	*Pelecanus occidentalis californicus*	Pacific coast from Canada to Mexico	Egg shell thinning due to pollutants such as DDT
California least tern	*Sterna albifrons brownii*	Pacific coast from S. San Francisco Bay, California, to S. Baja, California	Loss of nesting habitat (sandy beaches) due to increased human activity
American peregrine falcon	*Falco peregrinus anatum*	Coast and higher mountains inland	Egg shell thinning due to DDT; human disturbance
Southern bald eagle	*Haliaeetus leucocephalus leucocephalus*	Estuarine areas and inland around large lakes, reservoirs, and wetlands	Disturbance of nesting birds; illegal shooting; loss of nest trees; contamination of food chain by persistent pesticides
Light-footed clapper rail	*Rallus longirostris levipes*	Coastal salt marshes	Destruction of its natural habitat by filling for housing and industrial use, marine development, and water pollution destroying food species and/or habitat

[a]U.S. Department of the Interior, "Endangered and Threatened Wildlife and Plants," 41 F.R. 47180–47198.

Table 2.5. Endangered plant species of Orange and San Diego counties, California

Plant name[a]		Habitat and geography[a]
Scientific	Vernacular	
Acanthomintha ilicifolia	San Diego thornmint	Clay depressions on mesas and slopes; coastal sage scrub, chaparral; SW San Diego County
Arctostaphylos glandulosa var. *crassifolia*	Thickleaf manzantia	Sandy mesas and bluffs; chaparral; coast of San Diego County
Aster chilensis		Dry banks, grassy fields, etc., sea level to 5000 ft; many plant communities; mountains of San Diego County to Santa Barbara County
Astragalus tener titi	Coastal dunes rattleweed	Sandy places near the coast; coastal strand; near San Diego
Berberis nevinii	Nevin's bayberry	Sandy and gravelly places below 2000 ft; coastal sage scrub, chaparral; San Diego County
Brodiaea filifolia	Thread-leaved brodiaea	Heavy clay soil below 2000 ft; coastal sage scrub, chaparral; San Diego County
Brodiaea orcuttii	Orcutt's brodiaea	Near streams and around vernal pools and seeps, up to 5500 ft; chaparral; Yellow Pine Forest, San Diego County
Chorizanthe orcuttiana	Orcutt's chorizanthe	Sandy places; coastal sage scrub; San Diego County
Cordylanthus maritimus ssp. *maritimus*	Salt marsh bird's beak	Coastal salt marsh; Lower California to Oregon
Dicentra ochroleuca	Yellow dicentra	Occasional in dry disturbed places below 3000 ft; chaparral; Santa Ana and Santa Ynez mountains
Dichonda occidentalis	Western dichondra	Mostly dry sandy banks in brush or under trees; coastal sage scrub, chaparral, southern oak woodland; coastal San Diego and Orange counties
Dudleya multicaulis	Many-stemmed dudleya	Dry stony places below 2000 ft; coastal sage scrub, chaparral; San Onofre Mountain, Orange and San Diego counties
Dudleya stolonifera	Laguna Beach dudleya	Cliffs in coastal sage scrub; canyons near Laguna Beach, Orange County
Eryngium aristulatum var. *parishii*	San Diego coyote-thistle	Vernal pools; chaparral; San Diego region
Ferocactus viridescens	San Diego barrel cactus	Dry hills; coastal sage scrub, valley grassland; around San Diego, NW Lower California
Galium angustifolium ssp. *borregoense*		Creosote bush scrub; Borrego Valley, E. San Diego County
Githopsis filicaulis (last reported in 1884)	Mission Canyon blue-cup	Mission Canyon, San Diego
Hemizonia conjugans	Otay tarweed	Mesas; coastal sage scrub; SW San Diego County
Hemizonia floribunda	Tecate tarweed	Dry slopes and valleys below 3500 ft; coastal sage scrub, chaparral; S. San Diego County, N. Lower California
Limnathes gracilis var. *parishii*	Parish slender meadow-foam	Moist lake shores and wet places from 4500 to 5000 ft; Yellow Pine Forest; Cuyamaca and Laguna mountains
Monardella linoides ssp. *viminea*		Rocky washes below 1000 ft; coastal sage scrub, chaparral; SW San Diego County
Monardella macrantha var. *halli*	Hall's monardella	San Gabriel and San Bernardino mountains to Cuyamaca and Santa Ana mountains
Nolina interrata	San Diego nolina	Dry slope; chaparral; W. of Dehesa School, 8 miles east of El Cajon, San Diego County
Orcuttia californica var. *californica*	California orcuttia	Drying mud flats; valley grassland; San Diego County
Poa atropurpurea	San Bernardino bluegrass	Meadows and grassy slopes from 6000 to 7000 ft; Montane Coniferous Forest; San Diego County
Pogogyne abramsii	San Diego pogogyne	Beds of dried pools; chaparral, coastal sage scrub; mesas from San Diego to Miramar

[a]Nomenclature, habitat, and geography from P. A. Munz, *A Flora of Southern California*, University of California Press, Berkeley, Calif., 1974; and W. R. Powell, Ed., *Inventory of Rare and Endangered Vascular Plants of California*, Special Publication No. 1, Berkeley, Calif., 1974.

Source: U.S. Department of the Interior, "Endangered and Threatened Species, Plants," 41 F.R. 24542–24572.

(To convert ft to m, multiply by 0.3048.)

and (3) the Annual Report to the California Coastal Commission, August 1976-1977, by the Marine Review Committee,[28] a special study group established by the California Coastal Commission to estimate the consequences of operating SONGS 2 & 3. Because the ETS program contains the most recent data, included seasonal fluctuations in species occurrences and population densities, and evaluated the effects of SONGS 1 operation on the local marine environment, the description of the site aquatic ecology that follows is based on these data (obtained from November 1974 through December 1976). SONGS 2 & 3 are adjacent to SONGS 1, on the same site. Additionally, the effects of SONGS 1 operation are now a part of the environment of SONGS 2 & 3 and should therefore be included in a complete description of the site ecology.

The biotic communities relevant to an adequate description of the site ecology are the plankton, nekton, benthic, kelp, and intertidal communities.

2.5.2.1 Plankton

Bimonthly plankton sampling was conducted four times in 1975 and six times in 1976 at seven stations along the 10-m (33-ft) contour from 2.4 km (1.5 miles) upcoast to 6.7 km (4.2 miles) downcoast of the SONGS 1 intake/ discharge line (Fig. 2.3).

Phytoplankton

1975 Data. The 84 phytoplankton taxa recorded in the 1975 surveys are similar to those found in previous studies.[25] The phytoplankton was dominated numerically by dinoflagellates. *Prorocentrum micans* was the most abundant species, constituting 30 to 90% of the samples.[22] Other abundant organisms included *Prorocentrum* spp., *Ceratium* sp. A, and *Ceratium* sp. B. Several species of *Peridinium* and *Dinophysis* were also present. The number of taxa per station within each survey was relatively uniform. A complete list of phytoplankton taxa recorded during 1975 is given by station and survey in Appendix VIII, Table 2, p. 217 of ref. 25.

Chlorophyll α concentrations ranged from 0.24 to 2.32 mg/m^3 (0.004 to 0.04 grains/250,000 gal) during the four 1975 surveys.[25] Differences in chlorophyll α concentrations between stations were not significant. Differences were significant, however, between depths and between surveys; chlorophyll α concentrations were significantly greater at the 8-m (26-ft) depth, and the mean concentrations of September were significantly greater than those of the other survey months - May, July, and November.

Phaeopigment concentrations ranged from 0.08 to 1.23 mg/m^3 (0.076 to 0.174 grains/250,000 gal) during the four 1975 surveys.[25] Station differences were not significant, but differences in mean concentrations between surveys and between depths were significant. As with chlorophyll α, phaeopigment concentrations were greater at 8 m (26 ft) than at 1 m (3.3 ft), and the September survey showed the highest phaeopigment concentrations of all four surveys.

1976 Data. In 1976, 128 species or higher taxa of phytoplankton were reported from the six surveys conducted (Table II-2, pp. 11-13 of ref. 26). These taxa consisted of species when identifiable and higher taxa (genera, families, etc.) when identification to the species level could not be made. The taxa representing greater than 30% of any given sample by number were *Nitzschia* spp. (March and November), an unidentified pennate diatom (January, March, July, September, and November), *Gonyaulax* spp. (January and March), and *Prorocentrum micans* (May).[27]

Normal vertical distribution patterns were observed in 1976, as in 1975, with higher concentrations of chlorophyll α and phaeopigments again measured in the lower half of the 10-m (33 ft) water column. However, relatively high values of chlorophyll α were found during the January and May surveys in 1976, whereas in 1975, chlorophyll α concentrations were moderate in May and high in September. Also in contrast to 1975, there was no consistent vertical separation of diatoms from dinoflagellates.

Slightly higher surface temperatures at plankton stations nearest SONGS 1 during some surveys had no apparent effect on the distribution and abundance of phytoplankton; rather, distribution and abundance were apparently the result of natural spatial and temporal variation.[27]

Zooplankton

1975 Data. Zooplankton species encountered in the four 1975 surveys were common to the neritic waters of southern California.[22] A master species list of zooplankton found in the surveys is presented in Appendix VIII, Table 2, p. VIII-30 of ref. 22. The most common group consisted of copepodids of *Acartia* spp., usually accounting for more than 50% of the total number of individuals sampled.[22] Other species that commonly occurred in the samples were *Paracalanus parvus* copepodids, *Oikopleura* spp., *Evadne nordmanni*, *Labidocera trispinosa* copepodids,

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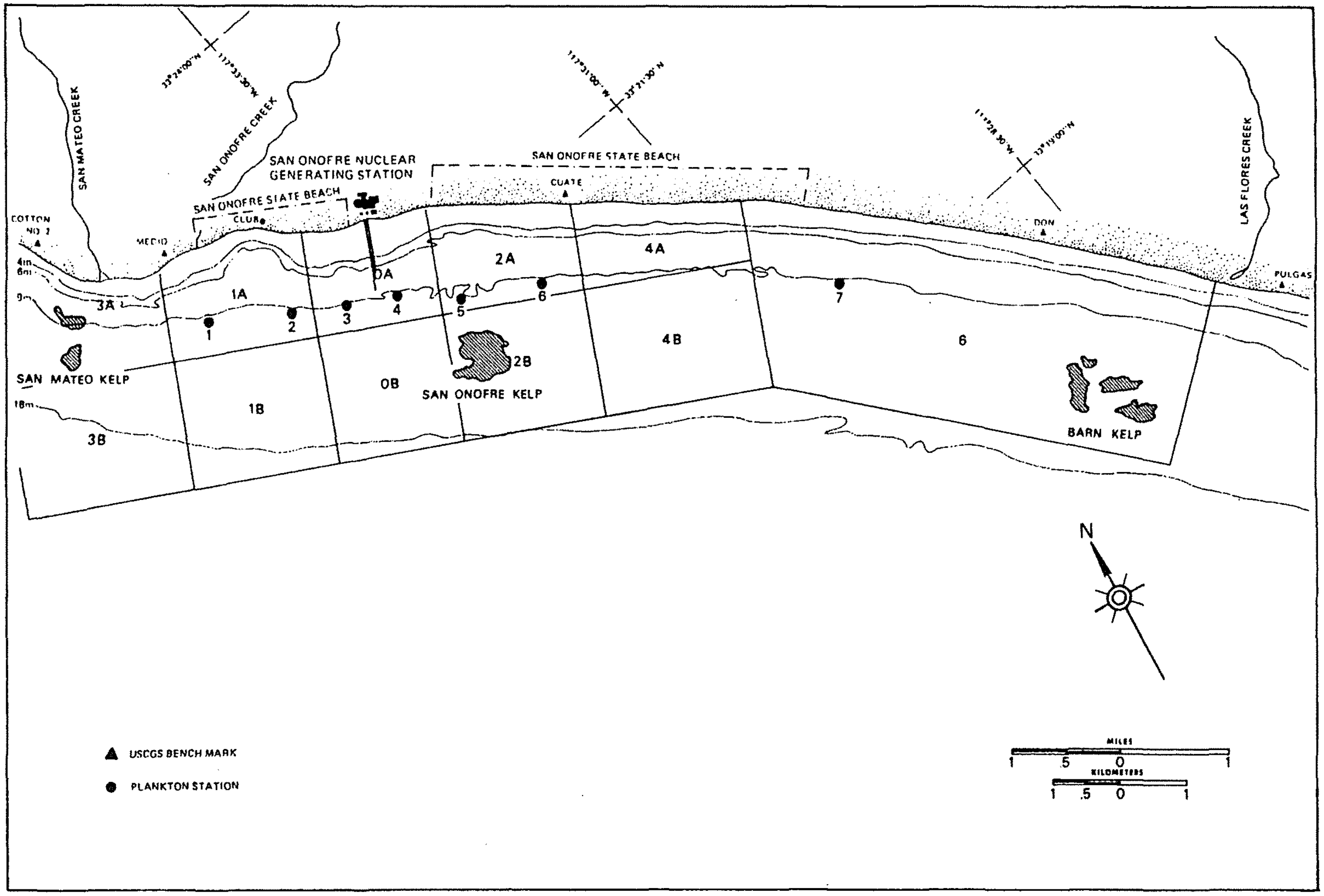

Fig. 2.3. Environmental Technical Specifications plankton station locations and environmental surveillance zones, San Onofre Nuclear Generating Station Unit 1. Source: Lockheed Center for Marine Research, *San Onofre Nuclear Generating Station Unit 1, Environmental Technical Specifications, Annual Operating Report, Vol. IV, Biological Data Analysis — 1976*, June 1977.

Sagitta euneritica, and Acartia tonsa. Less abundant species were adult Paracalanus parvus, cyphonautes larvae, Acartia clausi, and Clausocalanus spp. copepodids. Other species present usually accounted for less than 1% of any sample.[22]

Sampling stations were best differentiated by the distribution of five species: Sagitta euneritica, Corycaeus amazonicus, Oithona spp. copepodites, euphausiid larvae, and Podon polyphemoides. A clear separation of the stations, however, was not obtained, which suggests that no strong processes in the area acted to partition the environment.[25]

Total abundance per sampling station ranged from 600 to 10,900 per m^3 (568 to 10,322 per 250,000 gal) (Fig. 2.4), and total number of taxa ranged from 36 to 65 during the four surveys of 1975.[25] The number of taxa at station 4 near the SONGS 1 discharge was significantly higher than at all the other stations (Fig. 2.4).

1976 Data. In 1976, 115 species or higher taxa were reported from the six surveys performed (Table II-2, pp. 7-10 of ref. 26). Sixteen taxa were considered predominant because they were numerically dominant (number one in rank) during at least one survey, or because they represented more than 1% of the total number of individuals during the year.[27] These sixteen taxa constituted 90% of the total individuals recorded for the year.[27] The seasonal distribution of each of these taxa during the 1976 surveys is shown in Fig. 2.5. Significant differences were found among stations for all but five of the taxa, and significant differences were found between depths for all but six of them. All of these taxa exhibited significant differences among surveys.

Normal vertical distribution patterns were also observed in 1976, as in 1975, with higher concentrations of zooplankton observed in the lower half of the 10-m (33-ft) water column.

Although higher concentrations of zooplankton were measured near SONGS 1 in 1975, no effect of SONGS 1 was indicated by the 1976 studies. Even though water temperatures during the 1976 November survey (when SONGS 1 was off-line) were unusually warm for the season, the distribution and abundance of zooplankton, as with the phytoplankton, were apparently the result of natural spatial and temporal variation.[27]

2.5.2.2 Nekton

1975 Data

Quarterly nekton sampling was conducted in 1975 at six stations — three stations in the area of the SONGS 1 discharge (zone 0A) and three stations about 6706 m (22,000 ft) downcoast (zone 6) (Fig. 2.6). The downcoast stations (zone 6) acted as control areas not under the influence of the SONGS 1 discharge.

A total of 3206 individuals representing 49 species or higher taxa were taken during the four 1975 surveys.[25] The most abundant fish was the queenfish (Seriphus politus), which accounted for nearly twice the number of individuals in the year's catch than the second most abundant species. Other abundant fish were the walleye surfperch (Hyperprosopon argenteum), white croaker (Genyonemus lineatus), spotfin croaker (Roncador stearnsii), jacksmelt (Atherinopsis californiensis), and white surfperch (Phanerodon furcatus). Fourteen species were both abundant and common. Five of the 14 species displayed significant differences in their distributions between zones; four of these — jacksmelt, white seabass (Cynoscion nobilis), white croaker, and queenfish - were significantly more abundant in zone 0A, and the pile surfperch (Damalichthys vacca) was more abundant in zone 6.

The variability observed in abundance between zones was influenced significantly by the distribution of four species: white seabass, white croaker, white surfperch, and California corbina (Menticirrhus undulatus). The white seabass and white croaker were significantly more numerous in zone 0A, and the California corbina and white surfperch were significantly more numerous in zone 6.

The number of individuals and number of taxa also varied significantly among surveys. However, the degree of similarity of species composition within zones did not differ significantly from the degree of similarity between zones.

1976 Data

A taxonomic summary of the 1976 nekton sampling data by station and by survey can be found in Table III-4, pp. 17-18 of ref. 26. A total of 46 species was reported from these surveys. Seven species — queenfish, white croaker, white surfperch, walleye

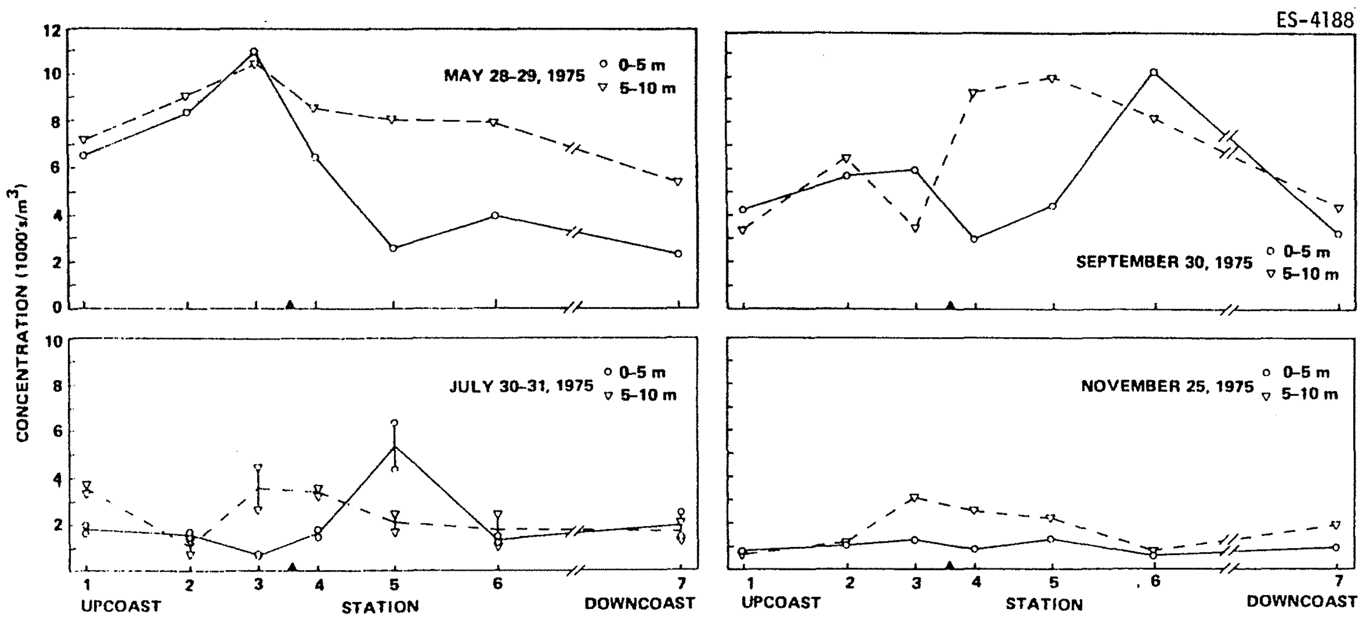

Fig. 2.4. Zooplankton concentrations from 1975 surveys. Open circles (o) and triangles (▽) indicate values from the upper and lower strata respectively. The relative distances of the plankton stations from SONGS 1 are shown. A solid triangle (▲) indicates the position of SONGS 1. A vertical bar connects the July replicates. Source: Lockheed Marine Biological Laboratory, *San Onofre Nuclear Generating Station Unit 1, Annual Analysis Report, Environmental Technical Specifications, January-December 1975*, 1976.

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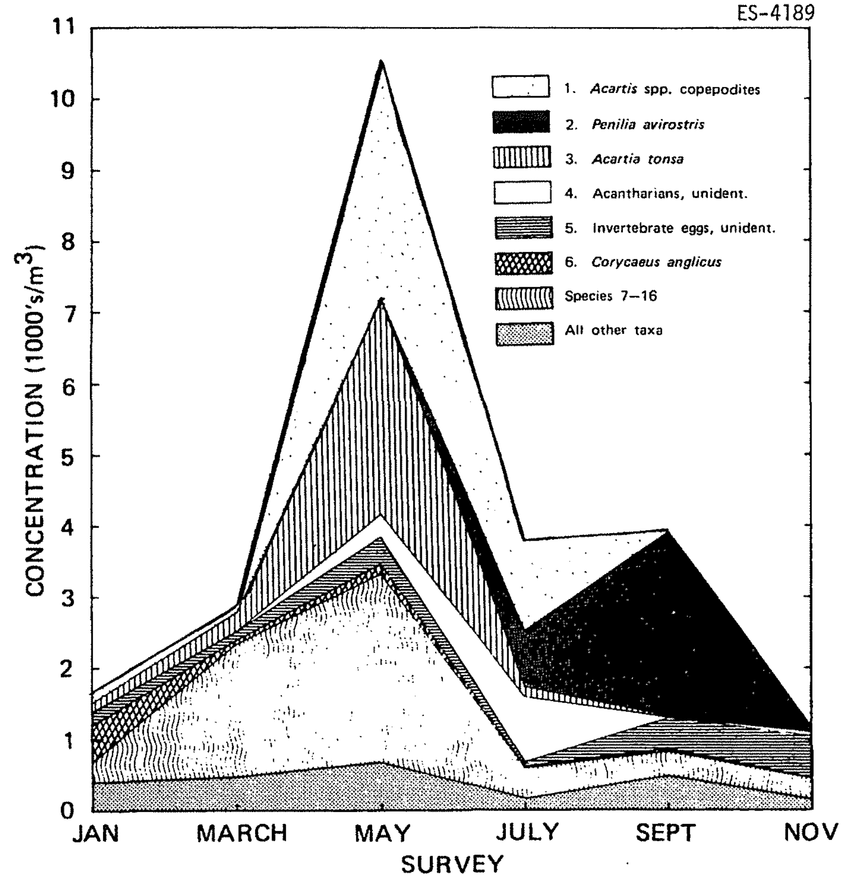

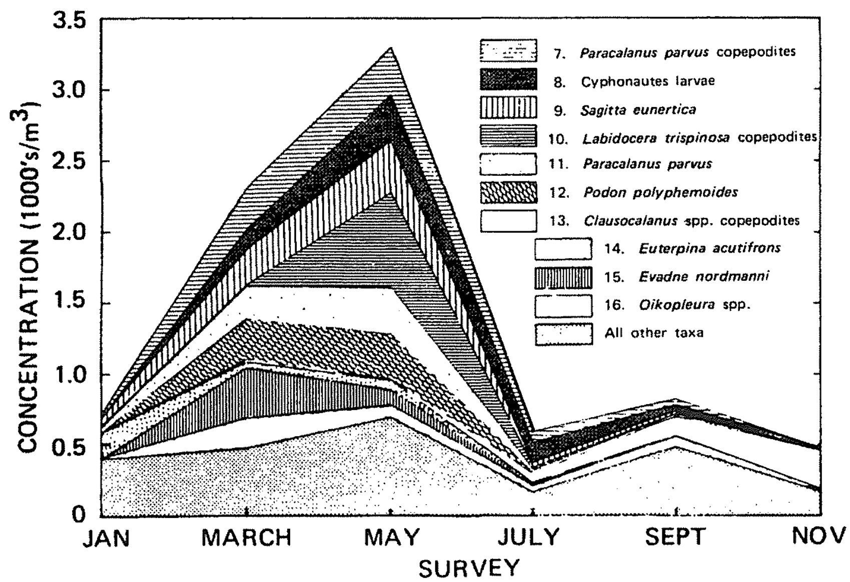

Fig. 2.5. Seasonal distribution of the 16 most abundant zooplankton taxa in 1976. Means of abundance during each survey are plotted. Source: Lockheed Center for Marine Research, *San Onofre Nuclear Generating Station Unit 1, Environmental Technical Specifications, Annual Operating Report, Vol. IV, Biological Data Analysis — 1976*, June 1977.

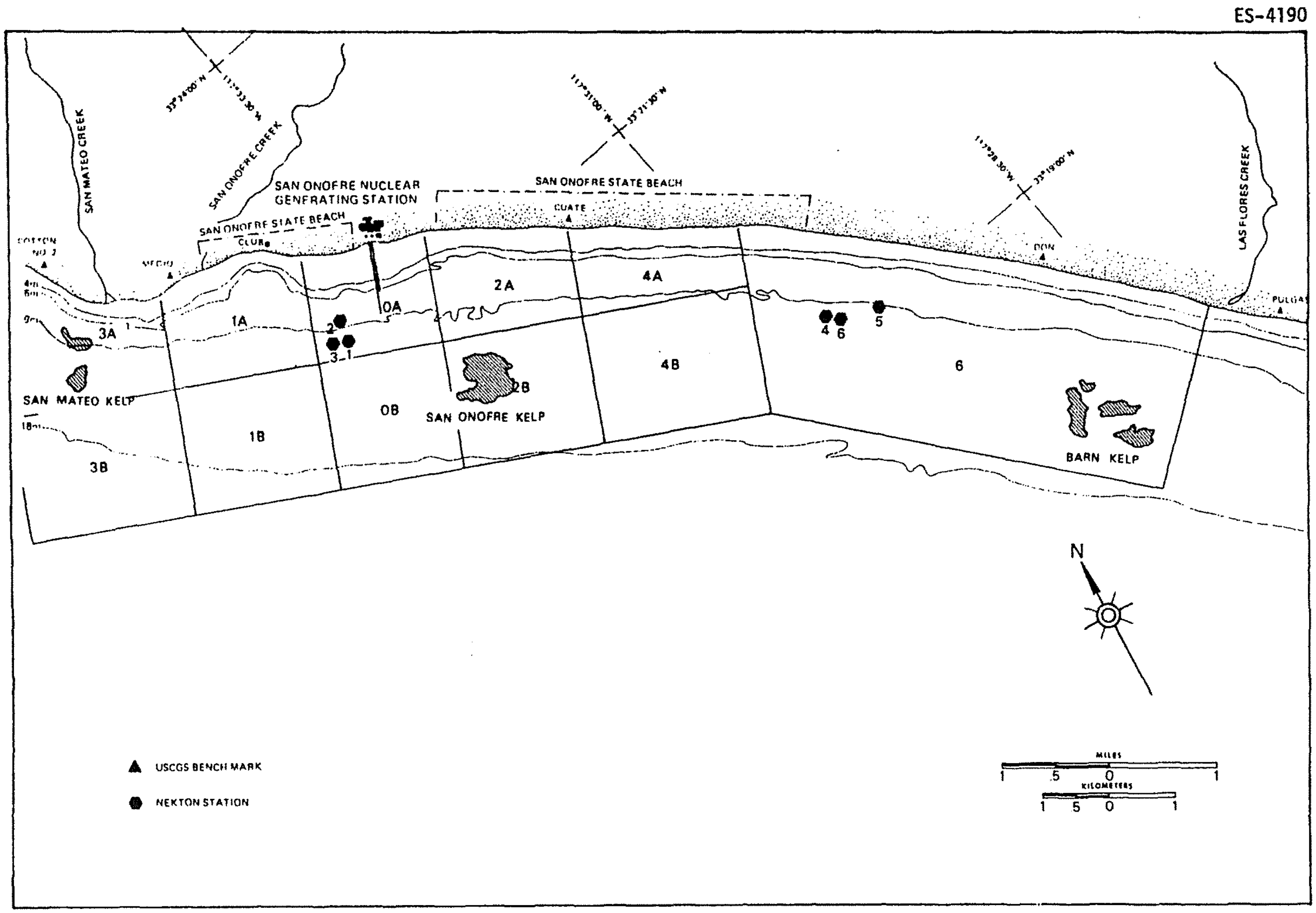

Fig. 2.6. Environmental Technical Specifications nekton station locations and environmental surveillance zones, San Onofre Nuclear Generating Station Unit 1. Source: Lockheed Center for Marine Research, *San Onofre Nuclear Generating Station Unit 1, Environmental Technical Specifications, Annual Operating Report, Vol. IV, Biological Data Analysis — 1976*, June 1977.

surfperch, black croaker (Chilotrema saturnum), spotfin croaker, and half moon (Medialuna californiensis) were captured in both zones.[27] As a group, these seven species accounted for 81.3% of the total catch for the year.[27] The first five of these species were tested for significant differences between zones and among surveys. Only the queenfish and white croaker showed a significant difference between zones, being significantly more abundant in zone 0A than in zone 6. The remaining three species did not differ significantly between zones.

In contrast, six predominant species in 1975 (bottom nets) contributed 82.3% of the individuals collected.[25] Of the predominant species netted in both years, only the queenfish and white croaker were significantly more abundant in zone 0A than in zone 6 during both years of the survey.

The spatial distribution of the queenfish, white croaker, and white surfperch differed significantly among the 1976 surveys. Temporally, the queenfish was found to be most abundant during the December survey and least abundant during the March survey. The white croaker was significantly more abundant during the December and March surveys than during the September and June surveys, and the white surfperch was significantly more abundant in the December catch than during all of the other 1976 surveys.

Significant differences were observed in the number of species between zones, with the number in zone 0A being significantly greater than the number in zone 6. Four species best discriminated between zones 0A and 6: white seabass, white croaker, yellowfin croaker (Umbrina roncador); and white surfperch.

There was also a significant difference among survey periods, with the number of species taken in March being significantly less than the number taken during all of the other surveys, which were not significantly different from each other.

The significant difference found in both number of individuals and number of species among surveys in 1976 was also found in 1975 although no obvious trend in species diversity was revealed (Fig. 2.7). On the other hand, a high similarity within zones existed during 1976; the 1975 data indicated similar but less distinct patterns.

The data suggest that the areas sampled in the two zones may support somewhat different nekton communities. Physical differences between the zones which may also affect the nekton results include the presence of the intake and discharge structures at SONGS 1 and riprap material in zone 0A, general differences in substrate type and composition between the zones, turbidity, and the presence of a dense stand of the phaeophyte Cystoseria spp. in the area of the zone 0A nekton stations. Temperature data collected during bimonthly cruises and nekton surveys revealed no obvious differences between zones, which indicates that temperature is not an important factor.

Fisheries statistics

Commercial and sport fisheries catch data for 1974 from the California Department of Fish and Game statistical blocks in the vicinity of SONGS 1 (Fig. 2.8) revealed that the number of fish per block ranged from 16,601 in block 737 to 123,246 in block 756.[27] With the exception of block 801, all of the blocks examined measured an increase in catch per unit effort between 1973 and 1974. However, the magnitude of the increase was small in comparison to the decrease shown by all of the blocks over the past 13 years.

The 1974 commercial catch reported a total of 46 taxa from the five blocks surrounding San Onofre.[27] The only taxon common to all five blocks was the Pacific bonito (Sarda chiliensis). Each of the five blocks yielded catches at about the expected level, based on the size of the blocks and the amount of coastline encompassed.[27]

2.5.2.3 Benthos

1975 Data

Three surveys conducted in 1975 at 11 benthic stations (Fig. 2.9) revealed a total of 160 species or higher taxa of epibenthic macrobiota (Tables X-1 to X-11, pp. X-12 to X-43 of ref. 22). The taxa represented members of 11 major taxonomic groups. Within zones not associated with kelp beds (zones 0A and 6), the flora was dominated by rhodophyte taxa throughout the year. Mollusks were the dominant fauna during April and October, whereas molluscan and chordate taxa occurred in similar numbers during the July sampling period. Rhodophytes were also the dominant floral component and mollusks were the dominant faunal component of the kelp bed biota at all kelp bed stations during all survey periods.

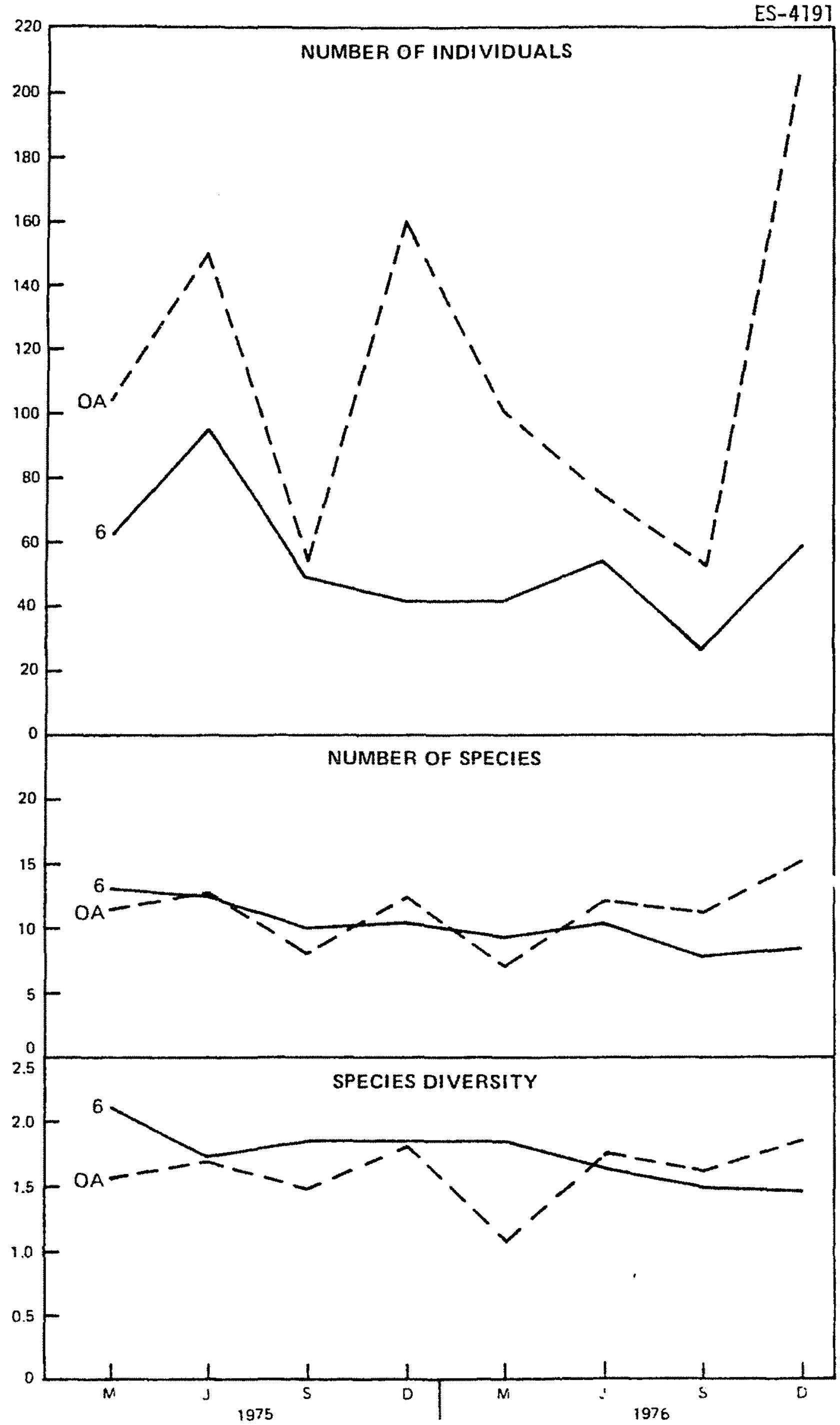

Fig. 2.7. The mean number of individuals and species per net by zone and species diversity of zones OA and 6 by survey during 1975 and 1976. Source: Lockheed Center for Marine Research, *San Onofre Nuclear Generating Station Unit 1, Environmental Technical Specifications, Annual Operating Report, Vol. IV, Biological Data Analysis — 1976,* June 1977.

The species whose distribution best discriminated between zones OA and 6 were the anthozoan Muricea californica, which occurred mostly in zone 6; the rhodophyte Prionitis spp., which was absent from zone 6; the holothuroid Parastichopus parvimensis, which occurred only in zone 6; and the gastropod Astrea undosa, which was observed only in zone OA.

The trophic composition based on the number of taxa of the two zones not associated with kelp beds (zones OA and 6) was similar among these zones and was dominated by suspension feeders and by primary producers during all surveys (Table 2.6).

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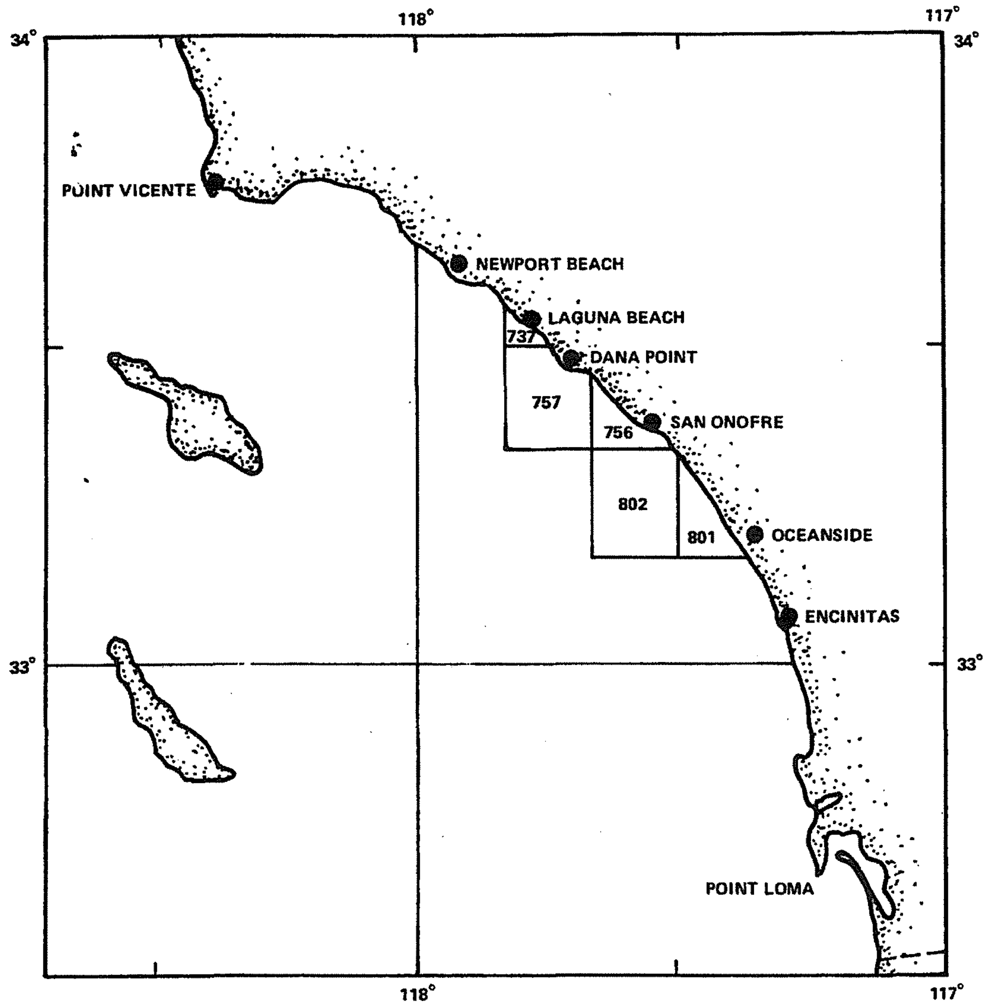

Fig. 2.8. California Department of Fish and Game catch statistic blocks in the vicinity of San Onofre. Source: Lockheed Center for Marine Research, *San Onofre Nuclear Generating Station Unit 1, Environmental Technical Specifications, Annual Operating Report, Vol. IV, Biological Data Analysis — 1976*, June 1977.

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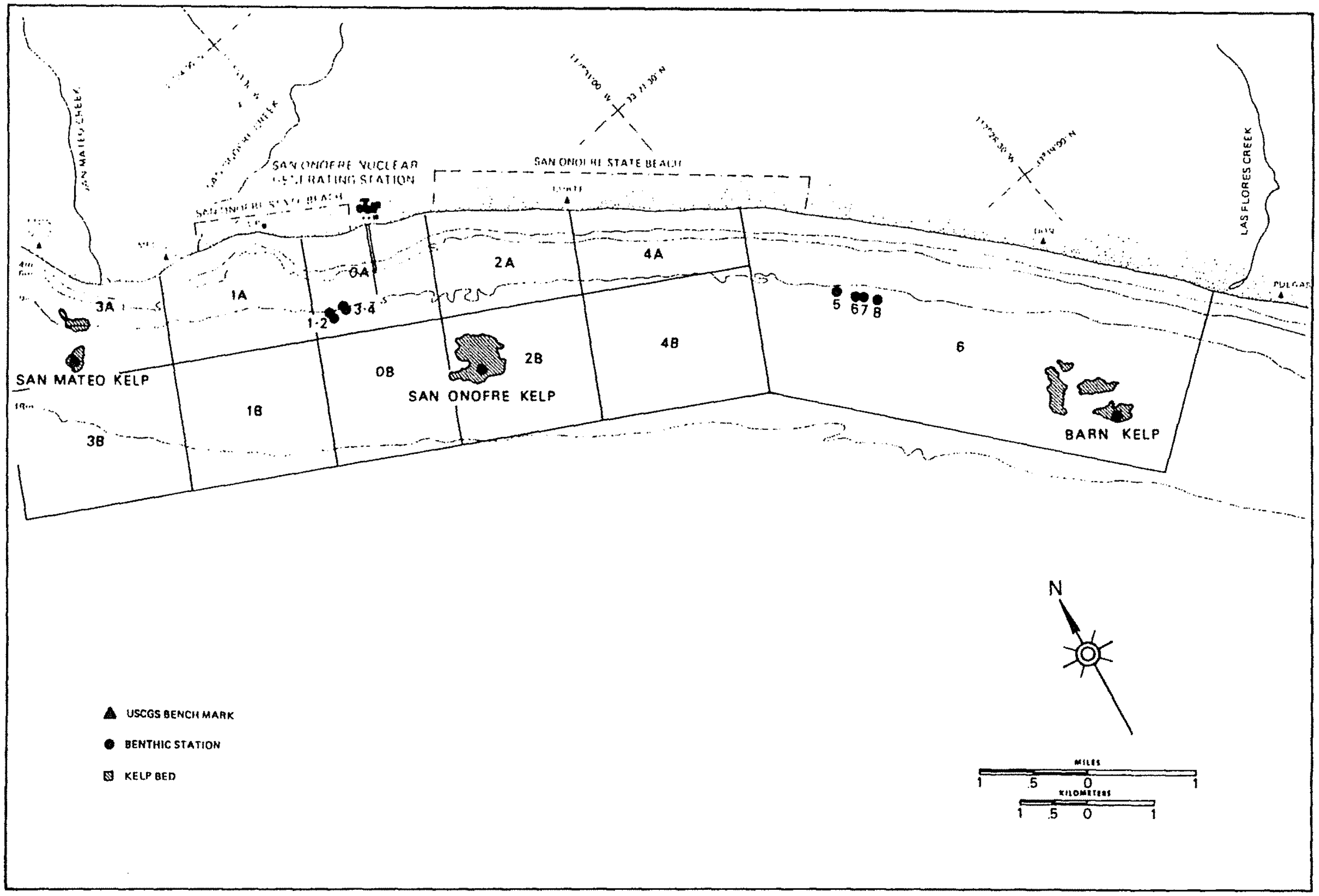

Fig. 2.9. Environmental Technical Specifications environmental surveillance zones, benthic station locations, San Onofre Nuclear Generating Station Unit 1. Source: Lockheed Center for Marine Research, *San Onofre Nuclear Generating Station Unit 1, Environmental Technical Specifications, Annual Operating Report, Vol. IV, Biological Data Analysis — 1976*, June 1977.

Table 2.6. Trophic composition (percent) of benthic taxa at discharge (zone 0A) and control (zone 6) based on the number of taxa of each trophic type present during 1975

Trophic types	April 10–18		July 15–18		October 13–17	
	Zone 0A	Zone 6	Zone 0A	Zone 6	Zone 0A	Zone 6
Primary producers	23	18	35	40	30	29
Suspension feeders	34	43	35	42	33	37
Grazers	10	3	12		12	5
Scavengers	13	13	7	10	12	11
Predators	20	22	12	7	13	18

Source: Lockheed Marine Biological Laboratory, *San Onofre Nuclear Generating Station Unit 1, Annual Analysis Report, Environmental Technical Specifications, January – December 1975,* 1976.

Kelp bed stations were best distinguished by four taxa: the gastropod *Cypraea spadicea*, which occurred only at San Onofre Kelp Bed; the anthozoan *Corynactis* spp., which occurred predominately at San Mateo and Barn kelp beds; the annelid *Spiochaetopterus costarum*, which did not occur at San Onofre Kelp Bed; and the white abalone, *Haliotis sorenseni*, which occurred only at San Onofre Kelp Bed. Twelve taxa were considered predominant at kelp bed stations: *Chelyosoma productum*, *Conus californicus*, *Corallina/Haliptylon*, *Corynactis* spp., Crustose corallines (unident.), *Dioptra* spp., *Leucilla nuttingi*, *Lytechinus pictus*, *Mitrella carinata*, *Muricea californica*, Pagurids (unident.), and *Rhodymenia* spp.

Trophic composition based on the number of taxa at the kelp bed stations was similar among stations and was dominated by suspension feeders (e.g., barnacles, which feed by filtering out suspended material) and primary producers (algae) during all surveys (Table 2.7).

Table 2.7. Trophic composition (percent) of benthic taxa at San Mateo (SMK), San Onofre (SOK), and Barn (BK) kelp beds based on the number of taxa of each trophic type present during 1975

Trophic types	April 10–18			July 15–18			October 13–17		
	SMK	SOK	BK	SMK	SOK	BK	SMK	SOK	BK
Primary producers	22	19	24	26	21	25	30	18	26
Suspension feeders	49	36	41	38	36	59	43	38	45
Grazers	2	17	9	8	12		7	10	4
Scavengers	12	12	9	12	12	9	7	12	10
Predators	15	17	17	16	18	6	12	22	16

Source: Lockheed Marine Biological Laboratory, *San Onofre Nuclear Generating Station Unit 1, Annual Analysis Report, Environmental Technical Specifications, January – December 1975,* 1976.

1976 Data

Diving surveys of the epibenthic macrobiota were conducted quarterly during 1976 at the same 11 benthic stations. A total of 159 species or higher taxa, which were members of 11 major taxonomic groups, were identified during the four surveys.[27] A taxonomic summary of these data by station and by survey is presented in Tables IV-1 and IV-2, pp. 21-28 of ref. 26. Zones 0A and 6 contained twelve predominant taxa whose combined abundance accounted for 84.3% of the total percent cover and 65.1% of the total enumerated individuals.[27] Seven of the twelve predominant taxa consisted of large taxonomic categories that were not field identifiable to a lower taxon. These seven taxa included parvosilvosa, unidentified ectoprocts, unidentified crustose coralline algae, and unidentified hydroids, rhodophytes, pelecypod siphons, and pagurids. These large taxonomic groups totaled 72% of the total percent cover and 20% of the total enumerated individuals for the entire year's data.[27] The magnitude of the abundances of these large taxonomic groups may be somewhat misleading, however, because each of these categories can contain members of several different species.[27]

The predominant taxa identified to at least the generic level consisted of *Rhodymenia* spp., *Bryopsis hypnoides*, *Diopatra ornata*, *Muricea californica*, and *Patiria miniata*. The distribution of these taxa among zones and stations is presented in Table V-12, p. 68 of ref. 27. The abundance of all of these taxa differed significantly between zones; *Rhodymenia* spp. and *Patiria miniata* were significantly more abundant in zone 0A, whereas *Bryopsis hypnoides*, *Diopatra ornata*, and *Muricea californica* were significantly more abundant in zone 6. None of these taxa differed significantly among surveys.

A greater degree of similarity in both species composition and abundance was found within zones than between zones. Distribution of the anthozoan *Murica californica* and the rhodophyte *Prionitis* spp. contributed the greatest to the differences between zones 0A and 6 in both years. Also in both 1975 and 1976, *M. californica* and the polychaete *Diopatra ornata* were significantly more abundant in zone 6. Species composition of the San Onofre Kelp Station was generally more similar to zone 0A stations than to the other kelp bed stations; this is much the same as the 1975 survey data.

No significant differences existed between zones or kelp bed stations in the distribution of taxa among trophic levels during 1975 or 1976.

Aerial infrared kelp survey

An aerial infrared kelp survey revealed that both Barn and San Onofre kelp beds showed a slight increase in total area during 1975 (Fig. 2.10). All of the kelp beds increased in size between February and May 1976 (Fig. 2.10). During the period May to September 1976, Barn and San Onofre kelp beds underwent an 80 and 92% decrease respectively.[27] At the time of the November 1976 survey, Barn Kelp Bed had increased to 77% of the area it had covered during the May survey, whereas San Onofre Kelp Bed again underwent a slight decrease.[27] San Mateo Kelp Bed remained essentially the same. The same general trends were encountered during mapping of the kelp beds by electronic positioning during 1975 and 1976 as part of the construction surveillance program for SONGS 2 & 3.

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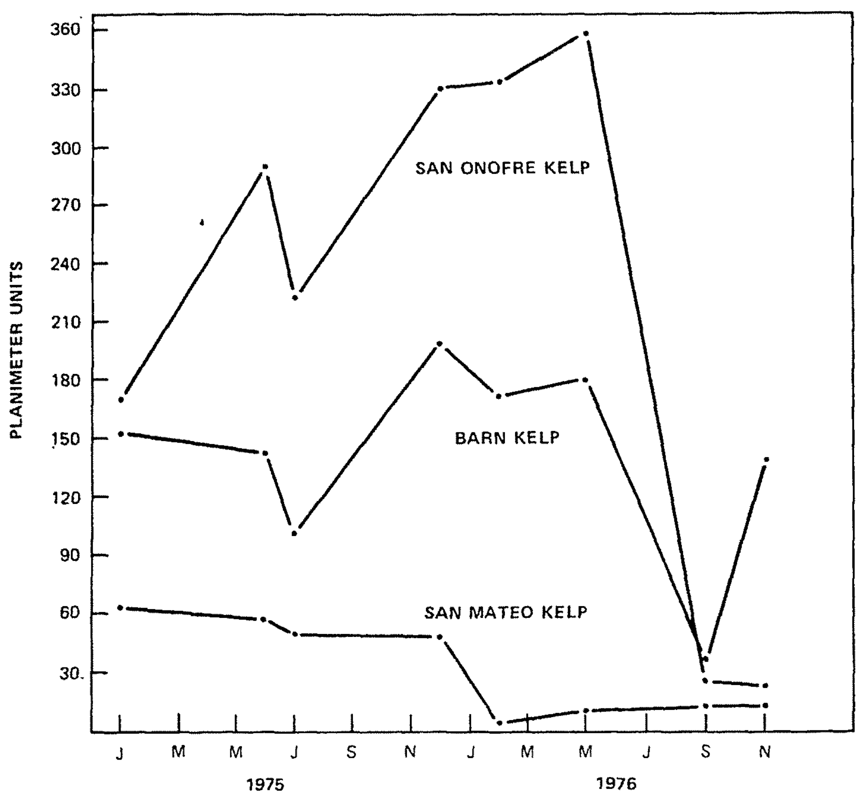

Fig. 2.10. Estimated relative total canopy area of San Mateo, San Onofre, and Barn kelp beds during 1975 and 1976, based on planimeter integration of aerial infrared photographs. Source: Lockheed Center for Marine Research, *San Onofre Nuclear Generating Station Unit 1, Environmental Technical Specifications, Annual Operating Report, Vol. IV, Biological Data Analysis — 1976*, June 1977.

Historical accounts of changes in kelp bed canopy areas throughout southern California have shown changes in magnitude equal to or much greater than those observed during this study, often over a short period of time.[27]

2.5.2.4 Intertidal community

1975 Data

During four intertidal surveys in 1975, 106 species or higher taxa representing 12 major taxonomic groups were observed at the five intertidal stations (Fig. 2.11).[25] These taxa are listed in Appendix XII, Tables 1 and 2, p. 246-52 of ref. 25. A comparison of the data collected in 1975 with historical data indicates that the fauna and flora encountered are typical inhabitants of this geographical area.[25] Phaeophytes, rhodophytes, and mollusks consistently exhibited the greatest number of taxa throughout the year at all stations. The distribution of five taxa were found to contribute significantly to the variability among stations: the rhodophytes *Corallina/Haliptylon*, *Pterocladia/Gelidium*, *Laurencia* spp.; the spermatophyte *Phyllospadix* spp.; and the anthozoan *Anthophleura* spp. Seventeen taxa, the majority of which were algae, were both common and abundant. The most abundant of these seventeen taxa were *Corallina/Haliptylon*, *Ulva* spp., and *Zonaria farolwii*.

Six predominant taxa exhibited distributions that varied significantly among stations, but no patterns that interrelated these differences were obvious. These six taxa were the anemone *Anthopleura* spp.; the rhodophytes *Corallina/Haliptylon*, *Lithothrix aspergillum*, *Pterocladia/Gelidium*; and the phaeophytes *Sargassum* spp. and *Zonaria farlowii*.

1976 Data

Quarterly intertidal sampling was also conducted in 1976. A taxonomic summary of these data by survey and station is presented in Table VI-1, pp. 35-38 of ref. 26.

Predominant taxa identified to at least the generic level were *Sargassum* spp., *Mitrella carinata*, *Macron lividus*, *Anthopleura elegantissima*, *Corallina/Haliptylon*, *Zonaria farlowii*, and *Dictyota/Pachydictyon*. The distribution of the abundance of these organisms for each station and for each survey is presented in Table VII-11, p. 104 of ref. 27. No significant differences were found in the abundance of *Dictyota/Pachydictyon*, *Macron lividus*, and *Mitrella carinata* among stations. The distribution of four taxa — *Corallina/Haliptylon*, *Zonaria farlowii*, *Sargassum* spp., and *Anthopleura elegantissima* — displayed statistically significant differences in abundance among stations. *Corallina/Haliptylon* was most abundant at station 5, *Zonaria farlowii* at stations 2 and 4, and *Sargassum* spp. was at station 3. The greatest number of *A. elegantissima* was observed at stations 1, 4, and 5.

The rhodophyte *Corallina/Haliptylon* contributed the most to the differences among stations during both 1975 and 1976 and was also predominant both years. During both years this taxon was more abundant at the station farthest downcoast of the SONGS 1 discharge and least abundant at the two stations upcoast of the discharge. Three other predominant taxa, *Sargassum* spp., *Zonaria farlowii*, and *Anthopleura elegantissima* exhibited statistically significant differences in abundance among stations during both 1975 and 1976. *Dictyota/Pachydictyon* exhibited no statistically significant differences in abundance among stations during either year.

No statistically significant difference in the distribution of taxa among trophic types existed among intertidal stations during either year. During both years, the intertidal communities of all stations were dominated by primary producers (algae).

The study area is accessible to considerable human intervention in the form of organism collecting in the tide pools, clam digging, surfing, and walking through intertidal cobble beds. Because of their accessibility via public roads, the stations nearest and upcoast of the generating station receive the heaviest use; the other stations receive less use because they are accessible only via hiking trail or the beach. Overall beach use in the study area is indicated by the San Onofre Beach State Park (which includes the study area) estimates of park use for 1976, which indicate that 378,483 people used the beach in the study area. The study area is also used heavily by clam diggers collecting littleneck clams, because this area is probably one of the most extensive and productive in the state. The large excavations and overturned cobble that result from clam digging may have considerable effect on the intertidal biota by disturbing habitats and interfering with mating activities.

Aerial infrared survey data on three occasions in 1976 revealed possible shore impingement of the 0.6°C (1°F) elevated temperature field at the four stations nearest the generating station. The 2°C (4°F) elevated field appeared to contact the shore immediately upcoast of the generating station but did not impinge on any intertidal cobble stations. Shore impingement of the elevated temperature field was not indicated in 1975.

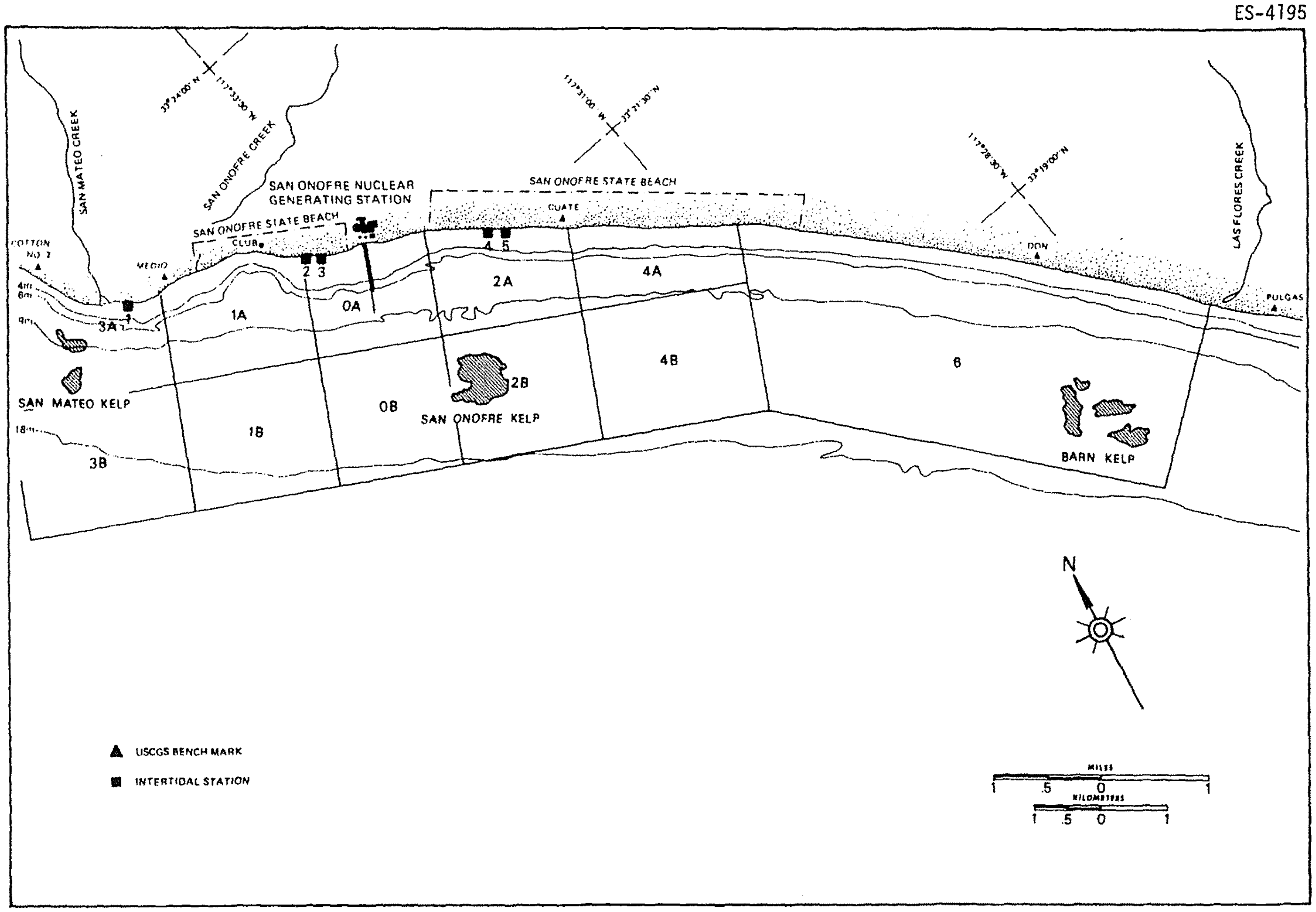

Fig. 2.11. Environmental Technical Specifications intertidal station locations and environmental surveillance zones, San Onofre Nuclear Generating Station Unit 1. <u>Source</u>: Lockheed Center for Marine Research, *San Onofre Nuclear Generating Station Unit 1, Environmental Technical Specifications, Annual Operating Report, Vol. IV, Biological Data Analysis — 1976,* June 1977.

Based on a comparison of the abundance of predominant taxa among stations and the similarity of stations during the study, the intertidal communities under study did not display a great deal of temporal variation during either 1975 or 1976. Minimal differences were detected among surveys with respect to the abundance of predominant taxa. These differences did not appear related to the offline condition of the generating station which occurred during two of four surveys.

2.6 BACKGROUND RADIOLOGICAL CHARACTERISTICS

The Environmental Protection Agency[29] has reported average background radiation dose equivalents for California as 96.6 millirems per person per year. The average background for San Diego is 104.6 millirems per person per year. (This is higher than the state average because of natural radioactivity in granitic rocks in the area.) Of the total for California, 42.2 millirems per person per year was attributed to cosmic radiation. Of this total external gamma radiation (primarily from K-40 and the decay products of the uranium and thorium series) was estimated at 36.4 millirems per person per year. The remainder of the whole body dose is due to internal radiation (mostly H-3, C-14, K-40, Ra-225, and Ra-228 and their decay products), which was estimated to average 18 millirems per person per year.

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***Available from NTIS only.

3. THE PLANT

3.1 RESUME

The domestic water supply and service water system will now be supplied by the Tri-Cities Municipal Water District rather than obtained from flash boilers as previously contemplated (Sect. 3.2.1). The major design changes that have environmental effects relate to the heat dissipation system. The revised heat dissipation system is described in Sect. 3.2.2. These revisions and others result in a change in the chemical effluents and are discussed in Sect. 3.2.4.1. Changes in the radioactive waste treatment systems are described in Sect. 3.2.3. Significant changes have occurred in the transmission lines; the revised transmission line system is described in Sect. 3.2.5.

3.2 DESIGN AND OTHER SIGNIFICANT CHANGES

3.2.1 Plant water use

Both fresh water and seawater will be used at SONGS 2 & 3. About 0.05 m^3/sec (1.65 cfs) of fresh water will be supplied by the Tri-Cities Municipal Water District for the domestic water supply system and service water system. The major portion of the domestic water requirement will be used for landscaping and associated functions. The service water system will provide water to miscellaneous systems and equipment throughout the operating areas. A large amount of this fresh water will be used at the intake screenwell area for cooling of pump bearings.

The source of seawater is the Pacific Ocean. Cooling water will be withdrawn from the ocean at a rate of 53.5 m^3/sec (1887 cfs). This water will be used for turbine plant cooling, component cooling, main condenser cooling, and for the fish handling system. The turbine plant and component cooling water systems are closed-cycle systems. Heat is transferred to the seawater by heat exchangers.

Further details of the plant water use are given in Fig. 3.1.

3.2.2 Heat dissipation system

Plant waste heat will be dissipated by means of a separate once-through cooling system for each unit. About 53.5 m^3/sec (1887 cfs) of seawater per unit is withdrawn from the ocean through a velocity-cap-type submerged intake, located about 975 m (3200 ft) from shore. The velocity cap is circular with a 15-m (50-ft) diameter. The lower lip of the cap is 2.7 m (9 ft) from the ocean bottom, and the interior separation of the upper and lower lip is 2.1 m (7 ft). The intake velocity will be about 0.5 m/sec (1.7 fps). The total water depth at the intake region is 9.1 m (30 ft). The intake structure is illustrated in Fig. 3.2.

Each unit has a Seismic Category I auxiliary intake structure to provide emergency core cooling. These structures are located approximately 32 m (100 ft) shoreward of the primary intake structures. Each structure has a 3.66 m (4-ft) ID vertical riser that extends upward from the intake conduit and is equipped with a velocity cap that is similar in design to that of the primary system. During normal operating conditions, water is estimated to enter the structure at 0.38 m/sec (1.3 fps). Details of these structures are shown in Fig. 3.2.

After passing through the intake, the cooling water for each unit will travel to the plant via a 5.5-m (18-ft) ID pipe that becomes a 4.9-m (16-ft) square box conduit at the shoreline. Here, water is delivered to a forebay leading to the intake structure screenwell. The water will then pass through a series of baffles as the channel widens to about 12.5 m (41 ft). At this point, the channel narrows and the main volume of water turns through an angle of 70°, where it passes through six adjacent sets of traveling bars and screens. A small volume of water does not turn towards these bars and screens but continues along the narrowing channel and enters the fish collection chamber.

Each screenwell is outfitted with traveling bar racks behind which are 1-cm (3/8-in.) mesh traveling screens. In the forebay behind the traveling screens are four 1/4-capacity vertical, wet pit,

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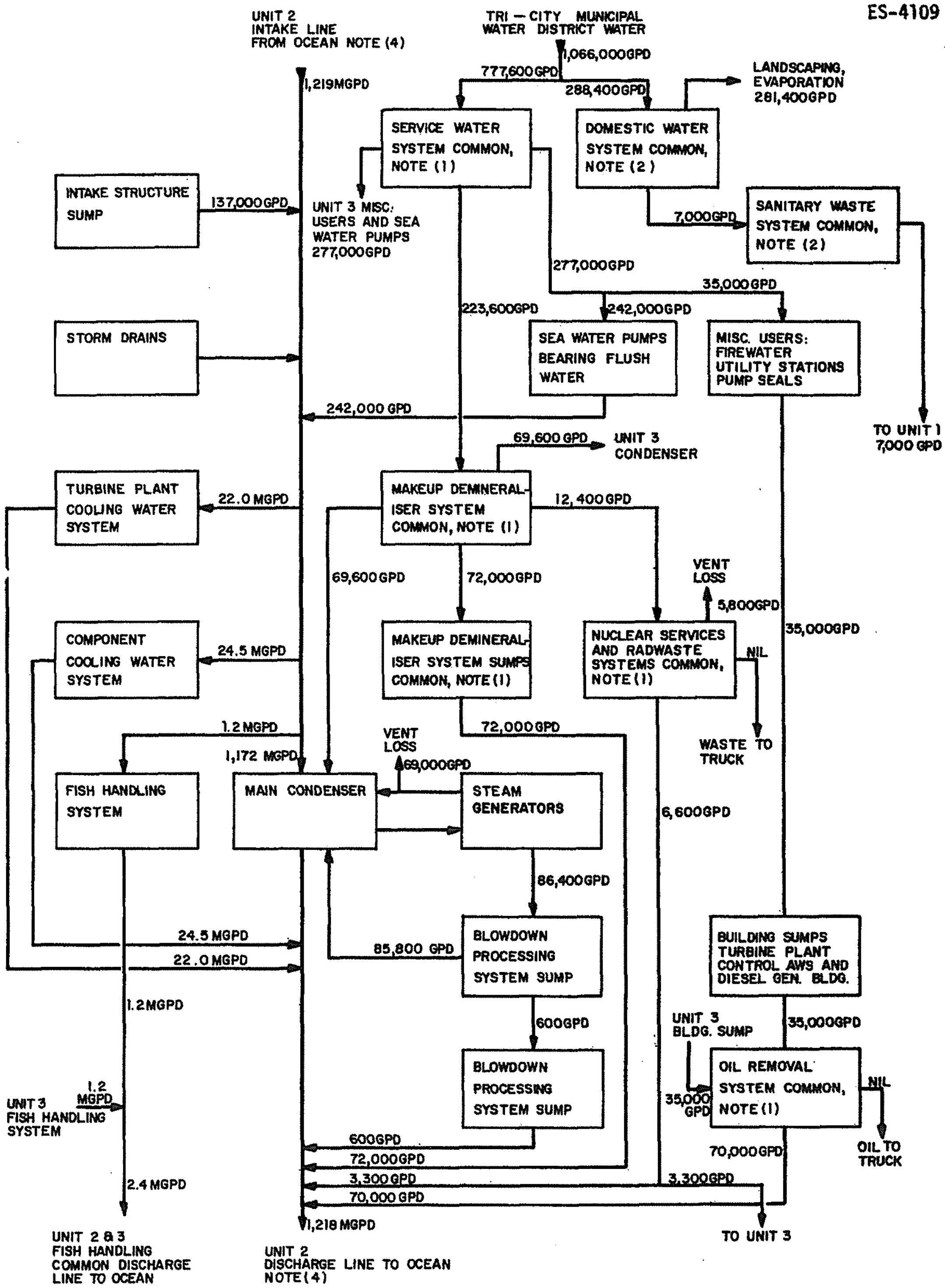

Notes: (1)Common system, serves Units 2&3
(2)Common system, serves Units 2&3, AWS bldg
(3)MGPD, millions of gallons per day
(4)Unit 3 flows are same as Unit 2
(5)To convert GPD to liters per day, multiply by 3.7854

Fig. 3.1. Plant water use. Source: ER, Fig. 3.3-1.

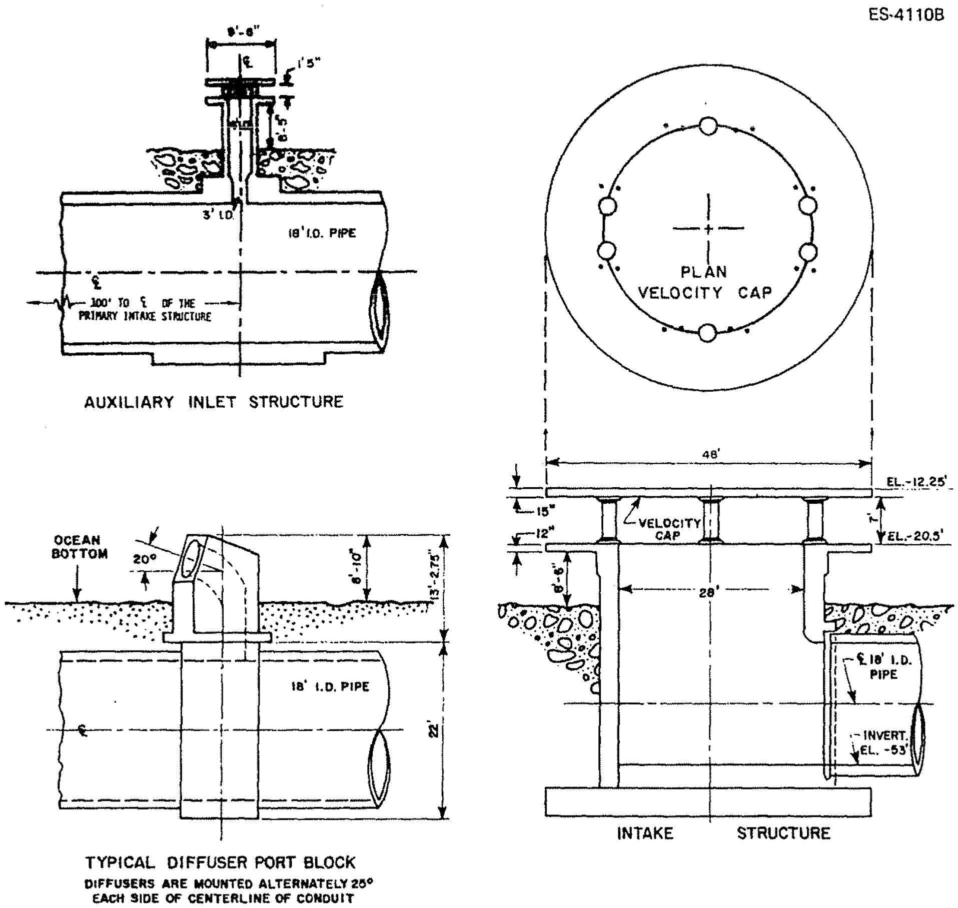

Fig. 3.2. Design details of the velocity-cap intake structure and typical diffuser port.
Source: ER, Fig. 3.4-2.
(To convert ft to m, multiply by 0.3048.)

circulating water pumps. These pumps provide 50.3 m^3/sec (1775 cfs) of water to a two-shelled condenser. This water experiences an 11.1°C (20°F) temperature rise across the condenser. About 2.1 m^3/sec (75.8 cfs) of water is withdrawn prior to reaching the condenser for use in the turbine plant cooling loop and the fish return systems. Details of the intake screenwell structure are shown in Fig. 3.3.

After passing through the condenser, the heated water will pass through the Amertap strainer, which collects the Amertap balls used for cleaning the condenser tubes. Subsequently, this heated water is supplemented by 1.1 m^3/sec (37.9 cfs) of water from the turbine plant cooling system and screenwashing. The water then passes into a seal well weir chamber designed to ensure proper siphon flow through the condenser. This chamber terminates into a 4.9-m (16-ft) square box conduit to which 1.1 m^3/sec (37.9 cfs) of nuclear component cooling water flow is added. At the shoreline, this square conduit joins a 5.5-m (18-ft) ID buried pipe that conveys the heated water to the diffuser.

The diffuser for each unit is about 762 m (2500 ft) in length, and each diffuser has 63 ports spaced 12 m (40 ft) apart. Each port extends 1.8 m (6 ft) from the bottom and is oriented from the horizontal at an angle of 20°. The ports are alternately aligned at angles of ±25° from the offshore direction. The port throat diameter will vary from 56 cm (22 in.) to 61 cm (24 in.), and the maximum discharge velocity from any port will be 4 m/sec (13 fps). The Unit 3 diffuser begins about 1150 m (3800 ft) from shore, and the Unit 2 diffuser begins about 1950 m (6400 ft) from shore. The Unit 2 diffuser is located about 220 m (722 ft) upcoast of the Unit 3 diffuser.

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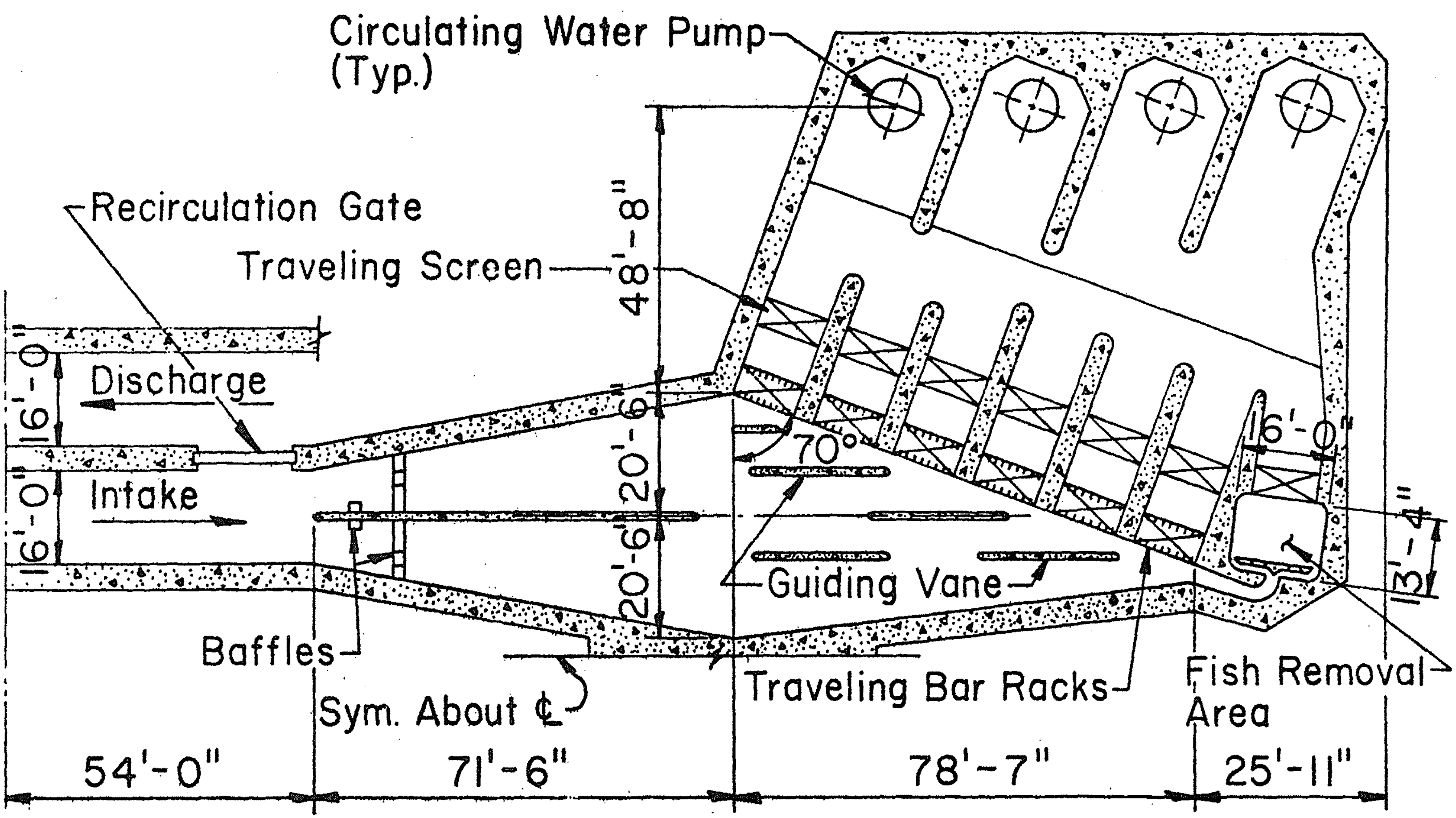

Fig. 3.3. Design details of the intake screenwell area. Source: ER, Fig. 3.4-3.
(To convert ft to m, multiply to 0.3048; to convert in. to mm, multiply by 25.4.)

3-4

To control biofouling, the circulating water system is designed to allow heated water to reach all portions of the system. To accomplish this, an intake/discharge crossover gate allows seawater to be drawn into the plant through the diffusers and the heated water to be discharged via the intake. To achieve the temperature required to control biofouling, each unit has a recirculation and crossover gate. This system allows the cooling water requirement to be reduced by recirculating a portion of the heated water through the condenser. The temperature rise will be proportional to the degree of recirculation. During diffuser heat treatment, the circulating water follows the normal path but with recirculation. Intake heat treatment is performed by opening the intake/discharge crossover gate to reverse the flow direction, as well as to allow recirculation. Circulating water flow paths for the various plant operations are shown in Fig. 3.4.

A fish return system minimizes the mortality of fish that have reached the intake screenwell area. The louvered bar racks are designed and oriented in such a way that the fish are encouraged to follow a narrowing channel terminating at a fish holding chamber. This chamber is equipped with a vertical elevator basket that periodically rises slowly from the bottom to capture the fish in the chamber. Subsequently, the fish are flushed from the basket with seawater into a 1.2-m (48-in.) diameter pipe, which returns them to the ocean via an offshore submarine outfall.

3.2.3 Radioactive waste systems

During the operation of SONGS 2 & 3 radioactive material will be produced by fission and by neutron activation of corrosion products in the reactor coolant system. From the radioactive material produced, small amounts of gaseous and liquid radioactive wastes will enter the waste streams. These streams will be processed and monitored within the station to minimize the quantity of radioactive nuclides ultimately released to the atmosphere and to the Pacific Ocean.

The waste handling and treatment systems to be installed at the station are discussed in the applicant's Final Safety Analysis Report (FSAR) and in the ER. Information submitted to meet the requirements of Appendix I to 10 CFR Part 50 is contained in both the FSAR and ER. In these documents, the applicant has presented an analysis of the radioactive waste treatment systems and has estimated the annual release of radioactive waste materials in liquid and gaseous effluents resulting from normal operation.

In the following paragraphs, the radioactive waste treatment systems are described, and an analysis is given based on the staff's model of the applicant's proposed radioactive waste treatment systems. The staff's model has been developed from a review of available data from operating nuclear power plants, adjusted to apply over a 30-year operating life. The reactor coolant activities and flow rates used in the staff's analyses are based on experience and data from operating reactors. As a result, the parameters used in the model and the calculated releases vary somewhat from those used in the applicant's evaluation.

On April 30, 1975, the NRC announced its decision in the rulemaking proceeding (RM 50-2) concerning numerical guides for design objectives and limiting conditions for operation to meet the criterion "as low as is reasonably achievable" for radioactive material in light-water-cooled nuclear power reactor effluents. This decision is implemented in the form of Appendix I to 10 CFR 50.[1] To effectively implement the requirements of Appendix I, the NRC staff has reassessed the parameters and mathematical models used in calculating releases of radioactive materials in liquid and gaseous effluents in order to comply with the Commission's guidance.

This guidance directed that current operating data, applicable to proposed radwaste treatment and effluent control systems for a facility, be considered in the assessment of the input parameters. These parameters, models, and their bases are given in NUREG-0017.[2]

By letter of February 25, 1976, the applicant was requested to submit additional information concerning the means proposed to keep levels of radioactive materials in effluents from SONGS 2 & 3 to unrestricted areas "as low as is reasonably achievable," in conformance with the requirements of Appendix I to 10 CFR 50. The applicant was also given the option of providing either a detailed cost benefit analysis or demonstrating conformance to the guidelines given in the September 4, 1975, Annex to Appendix I. The applicant chose to perform the cost-benefit analysis required by Sect. II.D of Appendix I to 10 CFR Part 50.

The staff performed an independent evaluation of the applicant's proposed methods to meet the requirements of Appendix I. The evaluation consisted of (1) a review of the information provided by the applicant, (2) a review of the applicant's proposed radwaste treatment and effluent control systems, (3) the calculation of new source terms based on models and parameters as given in NUREG-0017,[2] and (4) a cost-benefit analysis to determine the cost-effectiveness of proposed augments to the liquid and gaseous radwaste treatment systems.

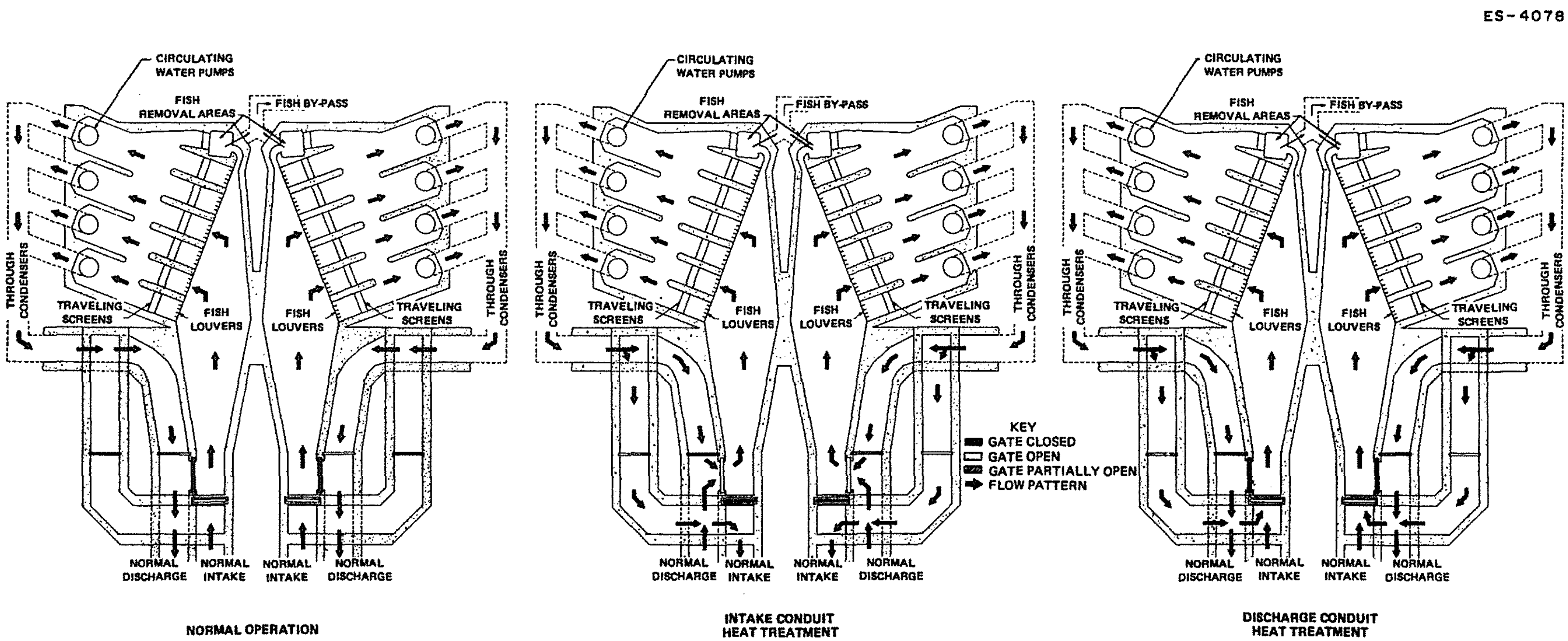

Fig. 3.4. Circulating water flow paths for normal plant operation, intake heat treatment, and discharge heat treatment. Source: Fig. 2-9 of *Thermal Effects Study Final Summary Report, San Onofre Generating Station Units 2 & 3 Volume 1*; Environmental Quality Analysts and Marine Biological Consultants, September 1973.

On the basis of the following evaluation, the staff concludes that the liquid and gaseous radio-active waste treatment systems for SONGS 2 & 3 are capable of maintaining releases of radioactive materials in liquid and gaseous effluents to "as low as is reasonably achievable" levels in accordance with 10 CFR Part 50.34a, and meet the requirements of Sect. II.A, II.B, II.C, and II.D of Appendix I to 10 CFR Part 50.[1]

3.2.3.1 Liquid radioactive waste treatment system

The liquid radioactive waste treatment system, which is shared by Units 2 and 3, will consist of equipment and instrumentation necessary to collect, process, monitor, recycle, or dispose of potentially radioactive liquid wastes generated during normal operation including anticipated operational occurrences. Liquid radioactive waste will be processed on a batch basis to permit optimum control of releases. Prior to release, samples will be analyzed to determine the types and amounts of radioactivity present; on the basis of the results, the waste will be recycled for reuse in the plant, retained for further processing, or discharged under controlled conditions to the Pacific Ocean via the circulating water outfall. A radiation monitor will automatically terminate liquid waste discharge if radiation measurements exceed a predetermined level in the discharge line. A schematic diagram of the liquid radioactive waste treatment system is given in Fig. 3.5.

The liquid radioactive waste treatment system will consist of the coolant radwaste (boron recovery) system, the miscellaneous (aerated) waste system, and the chemical waste system. The plant does not have a separate laundry and hot shower system; this function is combined in the aerated waste system.

The coolant radwaste system is shared by Units 2 and 3 and will process shim bleed and equipment drain wastes collected inside the reactor containment. The principal system components will be a gas stripper, four primary coolant radwaste holdup tanks, two preholdup demineralizers, two intermediate holdup tanks, two evaporator feed demineralizers, one evaporator, two polishing demineralizers, and two makeup storage tanks.

The miscellaneous liquid waste system will process non-reactor-grade liquid wastes, including floor drains, equipment drains containing non-reactor-grade water, and building sumps. After treatment these wastes will be transferred to the waste monitor tanks for reuse in the plant or for discharge to the Pacific Ocean via the circulating water outfall. The principal miscellaneous liquid waste system components will consist of one collection tank, four demineralizers, an optional evaporator, and two recycle monitor tanks. The liquid process stream may be routed through the optional evaporator if additional treatment is indicated.

The chemical waste system will process non-reactor-grade liquid wastes with high chemical content, including demineralizer regenerant solutions and laboratory drains. After treatment, these wastes will be transferred to the waste monitor tanks for reuse in the plant or for discharge to the Pacific Ocean via the circulating water outfall. The principal chemical waste system components will consist of one collection tank, an evaporator, two demineralizers, and two recycle monitor tanks.

The steam generator blowdown will be processed continually through a flash tank, with the liquid being cooled in a heat exchanger before passing through a filter and two demineralizers in series. The processed liquid is piped to the main condenser. The flashed steam is routed to the third point heater. The processed water will be reused in the plant, but may be discharged to the circulating water outfall under certain circumstances provided that radioactivity concentrations are below predetermined values.

Coolant radwaste system

Primary coolant will be withdrawn from the reactor coolant system at about 151 liters/min (40 gpm) and processed through the chemical and volume control system (CVCS). The letdown stream will be cooled, reduced in pressure, filtered, and processed through one of two mixed bed demineralizers. At the end of core cycle life this letdown stream will be passed through an anion demineralizer to remove boron when the feed and bleed mode of operation is not practicable. Radionuclide removal by the CVCS was evaluated by assuming 151-liters/min (40-gpm) letdown flow at primary coolant activity (PCA) through one mixed bed demineralizer (Li_3BO_3 form), and a continuous 30-liters/min (8-gpm) flow through one mixed bed demineralizer (H_3BO_3 form) for lithium control. The CVSC will be used to control the primary coolant boron concentration by diverting a side stream of about 3,785 liters/day (1000 gpd) per reactor of the treated letdown stream to the shared coolant radwaste system as shim bleed.

The shim bleed from the letdown stream will be processed through two mixed bed demineralizers (Li_3BO_3 form) in series, through a gas stripper, and routed to one of four 227,124-liter (60,000-gal) radwaste primary holdup tanks. Valve leakoffs and equipment drain wastes in the

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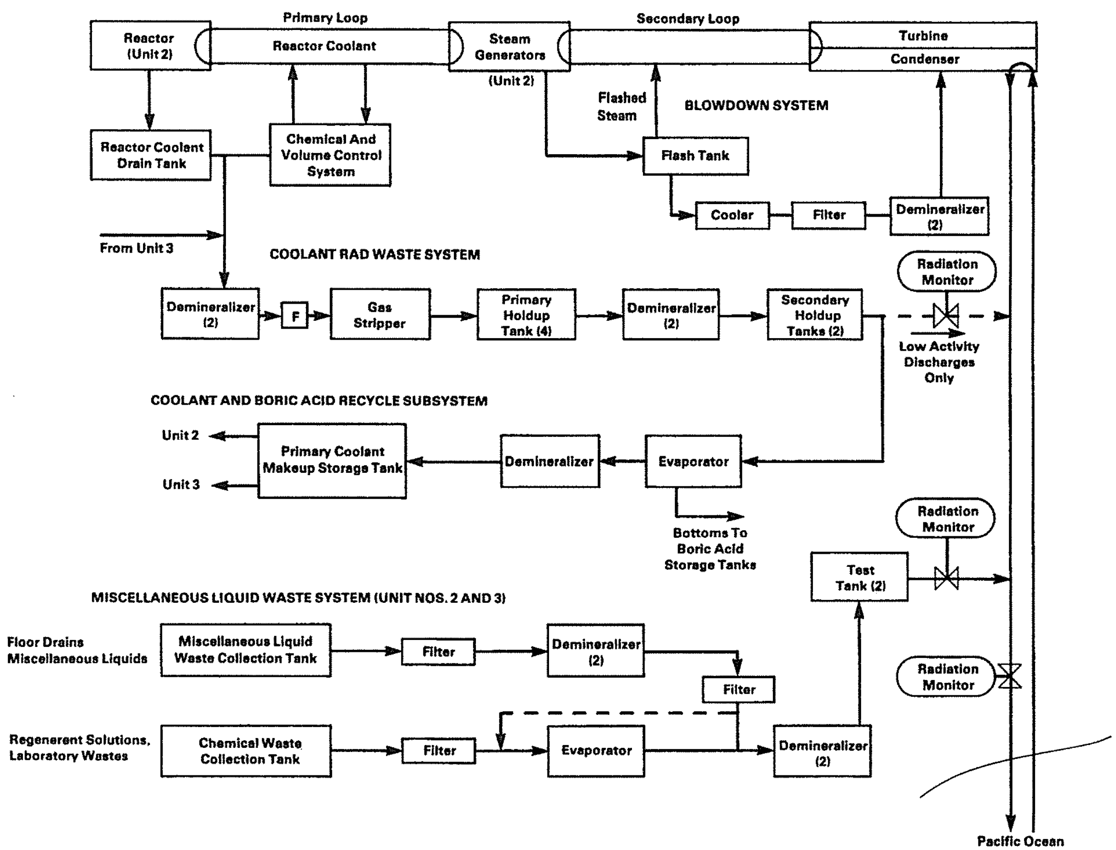

Fig. 3.5. SONGS 2 & 3 radioactive liquid waste treatment systems.

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reactor containment, as well as excess spent fuel pit water, will be processed as above and will be transferred to the radwaste primary holdup tanks where it will be combined with the shim bleed. These streams will form the inputs to the coolant radwaste system and will be processed batchwise from the four radwaste primary holdup tanks. The combined streams are next processed batchwise through two mixed bed demineralizers (H_3BO_3 form) and routed to one of two 454,248-liter (120,000-gal) radwaste secondary holdup tanks. From the radwaste secondary holdup tanks, the processed liquid can be recycled to the reactor coolant makeup tank, can be discharged to the circulating water outfall if radioactivity concentrations are within established limits, or can be processed further through a boric acid evaporator and mixed bed deborating and polishing demineralizers.

In the latter mode of operation, the boric acid recovered in the evaporator bottoms can be recycled. Because the system is capable of continuously operating in the boron recovery mode with inputs from both Units 2 and 3, and because the staff's source term calculation assumes a failed fuel rate of 0.12%, the staff's evaluation was made on the basis of the system being operated in the boron recycle mode. The staff calculated the collection time in a radwaste secondary holdup tank to be about 38 days, based on a combined input flow rate of 9463 liters/day (2500 gpd) from Units 2 and 3. Based on an assumption of 80% tank capacity and process flow rate of 189 liter/min (50 gpm), the staff calculated the decay time during processing to be about 1.3 days. If the radioactivity is below predetermined value, the treated stream may be pumped to the waste monitor release tank and discharged. The staff assumed that 10% of the treated stream will be discharged to the circulating water outfall and to the Pacific Ocean because of anticipated operational occurrences and for tritium inventory control. The decontamination factors listed in Table 3.1 were applied for radionuclide removal in the coolant radwaste system. The concentrated bottoms from the evaporator and the spent resins from the demineralizers will be transferred to the radioactive solid waste system for disposal by burial offsite.

Miscellaneous liquid waste system

The miscellaneous liquid waste system of the liquid radioactive waste treatment system is designed to collect and treat non-reactor-grade water for reuse within the plant from auxiliary building sumps, the containment sumps, and other miscellaneous sources. These wastes will be collected in a shared 22,712-liters (6000-gal) waste holdup tank at an input flow rate of about 5300 liters/day (1400 gpd) per unit. The staff calculated the collection time to be about 1.7 days. The wastes will be processed through four series connected mixed bed demineralizers and collected in a 94,635-liter (25,000-gal) test tank. The staff calculated the decay time during processing to be about 0.03 days. If necessary, the stream can be diverted to the evaporator in the chemical waste system for additional treatment.

The decontamination factors listed in Table 3.1 were applied for radionuclide removal in the miscellaneous liquid waste system of the liquid waste treatment system. The contents of the treated stream will be sampled periodically, recycled for further treatment, recycled for in-plant use, or discharged. The staff assumed that 100% of the treated stream will be released to the Pacific Ocean.

Evaporator bottoms and spent resins will be transferred to the radioactive solid waste system for disposal by burial offsite.

Chemical waste system

The chemical waste system of the liquid radioactive waste treatment system is designed to collect and treat non-reactor-grade liquid wastes from laboratory drains and from the regeneration of demineralizers. These wastes will be collected in a shared 94,635-liter (25,000-gal) chemical waste tank and sampled and analyzed. The wastes will be treated through the chemical waste system evaporator and two series connected mixed bed demineralizers prior to entering the waste monitor tanks. The staff calculated the collection time to be about 25 days, based on an input flow of about 1514 liters/day (400 gpd) per unit, and a decay time during processing of about 0.1 day.

Turbine building drain

The turbine building drains will be released through a radiation monitor to the Pacific Ocean via the circulating water outfall without treatment. The monitor will automatically terminate liquid discharge if radioactivity exceeds a predetermined level. The staff assumed a release of 27,255 liters/day (7200 gpd) per reactor and that the wastes will be discharged without processing.

Table 3.1. Principal parameters and conditions used in calculating releases of radioactive material in liquid and gaseous effluents from SONGS 2 & 3

Reactor power level, MWt	3600
Plant capacity factor	0.80
Failed fuel, percent	0.12[a]
Primary system:	
Mass of coolant, lb	5.6×10^5
Letdown rate, gpm	40
Shim bleed rate, gpd	1×10^3
Leakage to secondary system, lb/day	100
Leakage to containment building	*b*
Leakage to auxiliary building, lb/day	160
Frequency of degassing for cold shutdowns, per year	2
Secondary system	
Steam flow rate, lb/hr	1.5×10^7
Mass of liquid steam generator, lb	1.7×10^5
Mass of steam/steam generator, lb	1.2×10^4
Secondary coolant mass, lb	2.2×10^6
Rate of steam leakage to turbine building, lb/hr	1.7×10^3
Containment building volume, ft^3	2×10^6
Annual frequency of containment purges, shutdown	4
Containment low volume purge rate (cfm)	2000
Iodine partition factors, gas/liquid	
Leakage to auxiliary building	0.0075
Leakage to turbine building	1.0
Main condenser/air ejector, volatile species	0.15

Liquid radwaste system decontamination factors (DF)

	Coolant radwaste system (CRS)	Miscellaneous liquid-waste system	Chemical-waste system
I	1×10^5	1×10^3	1×10^4
Cs, Rb	2×10^5	2×10^1	1×10^5
Others	1×10^6	1×10^3	1×10^5

	All nuclides except iodine	Iodine
Radwaste evaporator DF	10^4	10^3
Coolant radwaste system evaporator DF	10^3	10^2

	Anions	Cs, Rb	Other nuclides
Boron recycle feed demineralizer DF, H_3BO_3	10	2	10
Primary coolant letdown demineralizer DF, Li_3BO_3	10	2	10
Evaporator condensate polishing demineralizer, H^+OH^-	10	10	10
Mixed-bed radwaste demineralizer	10^2(10)	2(10)	10^2(10)
Steam generator blowdown demineralizer	10^2(10)	10(10)	10^2(10)
Containment building internal recirculation system charcoal filter DF, iodine removal			10
Main condenser air-removal system charcoal bed DF, iodine removal			10

[a] This value is constant and corresponds to 0.12% of the operating power fission product source term as given in NUREG-0017 (April 1976).

[b] One percent per day of the primary coolant noble gas inventory and 0.001% per day of the primary coolant iodine inventory.

(To convert lb to kg, multiply by 0.4536; to convert gals to liters, multiply by 3.7854; to convert ft^3 to m^3, multiply by 0.0283.)

Steam generator blowdown

The steam generator blowdown system for Units 2 and 3 will continuously process steam generator blowdown at an average flow rate of 325,545 liters/day (86,000 gpd) per reactor (design flow rate is 1136 liters/min (300 gpm)). The blowdown from the two steam generators for each unit will be directed to a common flash tank. The liquid will be cooled, filtered, and treated through two series connected demineralizers before being returned to the main condenser. The flashed steam will be condensed in the main condenser hotwell. The staff did not consider any direct releases from this system to the environment.

Liquid waste summary

Based on the staff's evaluation of the radioactive liquid waste treatment systems and the parameters listed in Table 3.1, the staff calculated the release of radioactive materials in liquid waste effluent to be about 1.1 Ci per year per reactor, excluding tritium and dissolved gases. The staff estimates that about 300 Ci per year per reactor of tritium will be released to the Pacific Ocean. In comparison, the applicant estimated a release of radioactive material in liquid effluent, exclusive of tritium, to be about 0.67 Ci per year per reactor and a tritium release of 710 Ci per year per reactor. The differences between the staff's values and those of the applicant lie principally in assumptions as to the parameters used for each radwaste system component and the distribution of tritium between gaseous and liquid releases. The staff's calculations of the radionuclides expected to be released annually from SONGS 2 & 3 are given in Table 3.2.

On the basis of the calculated releases of radioactive materials in liquid effluents given in Table 3.2, the staff calculated the annual dose or dose commitment to the total body or to any organ of an individual in an unrestricted area, as shown in Table 5.3, to be less than 3 millirem per reactor and 10 millirem per reactor, respectively, in conformance with Sect. II.A of Appendix I to 10 CFR Part 50.

Cost-benefit analysis of liquid radwaste system augments

The staff evaluated potential liquid radwaste system augments based on a study of the applicant's system designs, the population dose information provided in Table 5.3 of this statement, a value of $1000 per total body man-rem and $1000 per man-thyroid-rem for reductions in dose by the application of augments, and the methodology presented in Regulatory Guide 1.110.[3]

The principal parameters used in this cost-benefit analysis are: (1) labor cost correction factor, FPC Region VIII, 1.2 (Regulatory Guide 1.110[3]); (2) indirect cost factor, 1.75 (Regulatory Guide 1.110[3]); (3) cost of money, 15%; and (4) capital recovery factor, 0.0806 (Regulatory Guide 1.110[3]).

The calculated total body and thyroid doses from liquid releases to the projected population within a 80 km (50-mile) radius of the station, when multiplied by $1000 per total body man-rem and $1000 per man-thyroid-rem, resulted in cost-assessment values of $170 per year per unit and $140 per year per unit respectively. Potential radwaste system augments were selected from the list given in Regulatory Guide 1.110.[3] The most effective augment was the optional use of an existing 0.189 liters/min (50-gpm) evaporator in the miscellaneous liquid waste system; however, the calculated total annualized cost of $80,000 for operation and maintenance of the augment exceeded the cost-assessment values of $170 per unit for the total body man-rem dose and $140 per unit for the man-thyroid-rem dose. The staff concludes, therefore, that there are no cost-effective augments to reduce the cumulative population dose at a favorable cost-benefit ratio, and that the proposed liquid waste management system meets the requirements of Sect. II.D of Appendix I to 10 CFR Part 50.

3.2.3.2 Gaseous radioactive waste treatment system

The gaseous radioactive waste treatement and building ventilation exhaust systems will be designed to collect, store, process, monitor, recycle, and/or discharge potentially radioactive gaseous wastes that will be generated during normal operation including anticipated operational occurrences. The system will consist of equipment and instrumentation necessary to reduce releases of radioactive gases and particulates to the environment.

The principal source of radioactive gaseous wastes are the gaseous waste processing system, condenser vacuum pump, and ventilation exhausts from the auxiliary, radwaste, fuel handling, containment, and turbine buildings. The principal system for treating gaseous wastes stripped from the primary coolant will be the gaseous waste processing system (GWPS). The GWPS will be a once-through nitrogen system containing a surge tank, two compressors, and six pressurized storage tanks. The off-gas from the main condenser air ejector will be processed through HEPA

Table 3.2. Calculated releases of radioactive materials in liquid effluents from SONGS 2 & 3

Nuclide	Curies per year per unit
Corrosion and activation products	
Cr-51	5.6(−4)
Mn-54	9(−5)
Fe-55	4.9(−4)
Fe-59	3(−4)
Co-58	4.8(−3)
Co-60	6.1(−4)
Np-239	2.5(−5)
Fission products	
Br-83	7(−5)
Rb-86	1.1(−3)
Rb-88	1.4(−2)
Sr-89	1(−4)
Sr-91	4(−5)
Y-91m	3(−5)
Y-91	2(−5)
Zr-95	2(−5)
Nb-95	1(−5)
Mo-99	1.9(−2)
Tc-99m	1.5(−2)
Ru-103	1(−5)
Rh-103m	1(−5)
Te-127m	8(−5)
Te-127	1.1(−4)
Te-129m	4.1(−4)
Te-129	2.8(−4)
I-130	1.9(−4)
Te-131m	4(−4)
Te-131	7(−5)
I-131	8.1(−2)
Te-132	6.2(−3)
I-132	7.8(−3)
I-133	5.3(−2)
I-134	2.3(−4)
Cs-134	3.5(−1)
I-135	9.5(−3)
Cs-136	1.7(−1)
Cs-137	2.5(−1)
Ba-137m	1.6(−1)
Ba-140	6(−5)
La-140	4(−5)
Ce-141	2(−5)
Pr-143	1(−5)
All others	5(−5)
Total, except H-3	1.1
H-3	300

filters and charcoal absorbers prior to release to the environment. The containment building atmosphere will be recirculated through HEPA filters and charcoal absorbers prior to release to the environment. Ventilation exhaust air from the auxiliary building and the fuel handling area will not be processed prior to release to the environment. The turbine building ventilation exhaust air will be released to the environment without treatment. The gaseous waste and ventilation treatment systems are shown schematically in Fig. 3.6.

Gaseous waste processing system (GWPS)

The GWPS will be designed to collect and process gases stripped from the primary coolant in the CVCS, coolant radwaste system, and miscellaneous tank cover gases. The GWPS is shared between Units 2 and 3. The GWPS will contain an inventory of nitrogen and hydrogen which will act as a carrier gas to transport radioactive gases removed from the primary coolant. Hydrogen and nitrogen cover gases from the volume control and reactor coolant drain tanks, and gases stripped in the coolant radwaste system degasifier will be collected, compressed, and stored in one of six pressurized storage tanks. The storage tanks will collect and store gases to allow short-lived radionuclide decay. After holdup, the gases will be discharged to the environment.

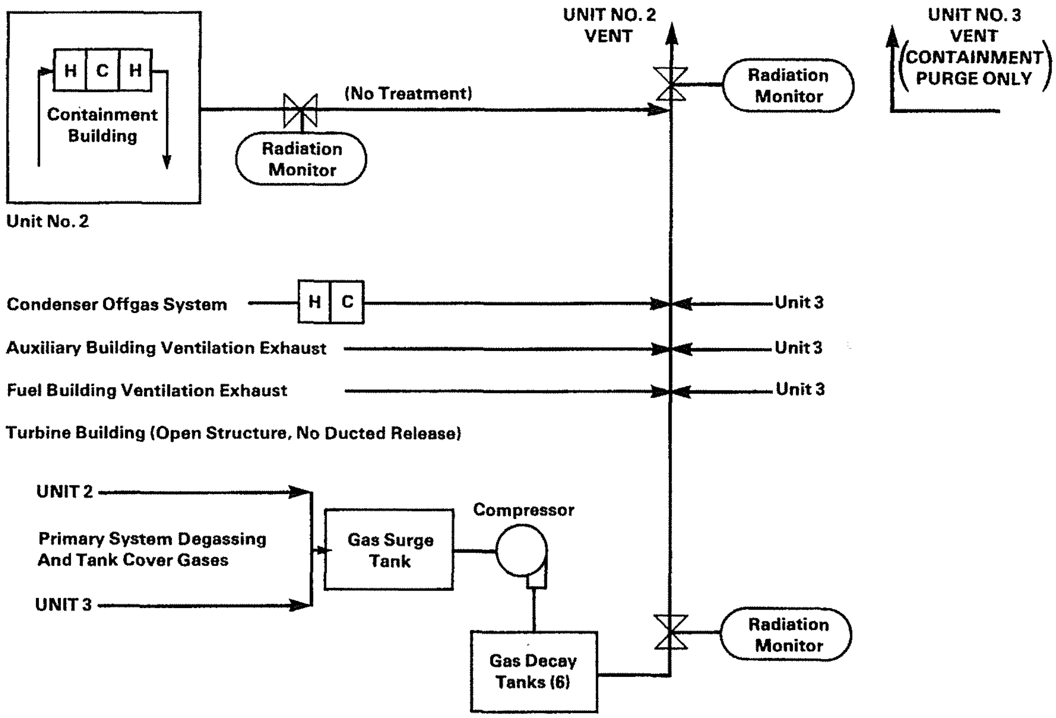


Fig. 3.6. SONGS 2 & 3 radioactive gaseous waste treatment systems.

In its evaluation, the staff assumed three tanks for storage, with two tanks held in reserve for back-to-back shutdowns, and one tank in the process of filling. Each tank has a volume of 14.16 m^3 (500 ft^3) and operates at 300 psig. On this basis, the staff calculated a holdup time of 90 days prior to discharge of gases to the environment.

Containment ventilation system

Radioactive material will be released inside the containment when primary system leakage occurs. The staff assumed on the basis of system parameters that the containment will be purged continuously during power operations at 56.6 m^3/min (2000 cfm) and in addition will have four high volume shutdown purges per year at 1132 m^2/min (40,000 cfm). Prior to purging, the containment atmosphere will be recirculated through HEPA filters and charcoal absorbers. The staff assumed radionuclide removal during the recirculation phase to be based on a flow rate of 453 m^3/min (16,000 cfm), system operation for 16 hr, a mixing efficiency of 70%, a particulate decontamination factor of 100 for HEPA filters, and an iodine decontamination factor of 10 for charcoal absorbers. The purge exhaust gases are released without filtration or other treatment.

Ventilation releases from other buildings

Radioactive materials will be released into the plant atmosphere due to leakage from equipment transporting or handling radioactive materials. Ventilation air from the auxiliary building and fuel building is not processed prior to release. The staff estimated that 72.58 kg (160 lb) of primary coolant per day will leak to the auxiliary building with an iodine partition factor of 0.0075. Small quantities of radionuclides will be released to the open turbine building, based on an estimated 771 kg/hr (1700 lb/hr) of steam leakage. The open turbine building releases will be released directly to the environment.

Main condenser air ejector

Off-gas from the main condenser air ejectors will contain radioactive gases as a result of primary to secondary leakage. In its evaluation, the staff assumed a primary to secondary leak rate of 45 kg/day (100 lb/day). Noble gases and iodine will be contained in steam generator leakage and released to the environment through the main condenser air ejectors in accordance with the partition factors listed in Table 3.1. The air ejector exhaust will be released to the environment through HEPA filters and charocal absorbers.

Gaseous waste summary

Based on the staff's evaluation of the gaseous radioactive waste treatment and building ventilation systems and the parameters listed in Table 3.1, the staff calculated the release of radioactive materials in gaseous effluents to be about 15,000 Ci per year per unit for noble gases and 0.44 Ci per year per unit for iodine-131. In comparison, the applicant estimated a release of 8600 Ci per year per unit for noble gases and 0.096 Ci per year per unit for iodine-131. The staff estimated a release of 0.39 Ci per year per unit of particulates and 1100 Ci per year per unit of tritium. The applicant estimated a release of 0.2 Ci per year per unit of particulates and 710 Ci per year per unit of tritium.

The staff's calculated annual releases of radioactive materials in gaseous effluents from radionuclides expected to be released annually from SONGS 2 & 3 are given in Table 3.3. Based on the calculated releases of radioactive materials in gaseous effluents given in Table 3.3, the staff calculated the annual air in an unrestricted area, as shown in Table 5.3. to be less than 10 millirads per reactor for gamma radiation or 20 millrads per reactor for beta radiation and the annual external doses to the total body and skin of an individual in an unrestricted area to be less than 5 millirems and 15 millirems, respectively, and an organ dose of less than 15 millirems per reactor for radioiodine and radioactive particulates in conformance with Sect. II.B and II.C of Appendix I to 10 CFR 50.

Table 3.3. Calculated releases of radioactive materials in gaseous effluents from SONGS 2 & 3 (Curies per year per unit)

Nuclide	Decay tanks	Reactor building	Auxiliary building	Turbine building	Air ejector	Total
Kr-83m	*a*	2	*a*	*a*	*a*	2
Kr-85m	*a*	24	2	*a*	2	28
Kr-85	430	170	5	*a*	3	610
Kr-87	*a*	5	1	*a*	*a*	6
Kr-88	*a*	30	4	*a*	3	37
Kr-89	*a*	*a*	*a*	*a*	*a*	*a*
Xe-131m	*a*	90	3	*a*	2	95
Xe-133m	*a*	140	5	*a*	3	150
Xe-133	*a*	13,000	410	*a*	260	14,000
Xe-135m	*a*	*a*	*a*	*a*	*a*	*a*
Xe-135	*a*	120	8	*a*	5	130
Xe-137	*a*	*a*	*a*	*a*	*a*	*a*
Xe-138	*a*	*a*	*a*	*a*	*a*	*a*
Total noble gases						15,000
I-131	*a*	0.35	0.08	0.0042	0.005	0.44
I-133	*a*	0.27	0.09	0.0033	0.0056	0.37
Mn-54	4.5(−3)[b]	2.2(−2)	1.8(−2)	*c*	*c*	4.4(−2)
Fe-59	1.5(−3)	7.4(−3)	6(−3)	*c*	*c*	1.5(−3)
Co-58	1.5(−2)	7.4(−2)	6(−2)	*c*	*c*	1.5(−2)
Co-60	7(−3)	3.3(−2)	2.7(−2)	*c*	*c*	6.7(−2)
Sr-89	3.3(−4)	1.7(−3)	1.3(−3)	*c*	*c*	3.3(−3)
Sr-90	6(−5)	2.9(−4)	2.4(−4)	*c*	*c*	5.9(−4)
Cs-134	4.5(−3)	2.2(−2)	1.8(−2)	*c*	*c*	4.4(−2)
Cs-137	7.5(−3)	3.7(−2)	3(−2)	*c*	*c*	7.4(−2)
Total particulates						1.2
H-3						1,100
C-14	7	1	*a*	*a*	*a*	8
Ar-41	*a*	25	*a*	*a*	*a*	25

[a] Less than 1 Ci/year for noble gases and carbon-14, less than 10^{-4} Ci/year for iodine.
[b] Exponential notation: 4.5(−3) = 4.5×10^{-3}.
[c] Less than 1% of total for this nuclide.

Cost-benefit analysis of gaseous radwaste system augments

The staff has evaluated potential gaseous radwaste system augments based on a study of the applicant's system designs, the population dose information provided in Table 5.3 of this statement, a value of $1000 per total body man-rem and $1000 per man-thyroid-rem for reductions in dose by the application of augments, and the methodology presented in Regulatory Guide 1.110.[3]

The calculated total body and thyroid doses from gaseous releases to the population within a 80 km (50-mile) radius of the station, when multiplied by $1000 per total body man-rem and $1000 per man-thyroid-rem, resulted in cost-assessment values of $21,000 per year per unit and $46,000 per year per unit respectively. Potential radwaste system augments were selected from the list given in Regulatory Guide 1.110. The most effective augment considered was the installation of charcoal adsorbers and HEPA filters on the containment mini-purge ventilation exhaust. The addition of this augment would result in a dose reduction of approximately 6.3 total-body man-rem and 23.8 thyroid man-rem with corresponding cost assessment values of $6,300 and $23,800, respectively. The calculated total annualized cost of $26,500 for the augment is more than the annual cost assessment values of $6,300 and $23,800 given above. The staff concludes, therefore, that there are no cost-effective augments to reduce the cumulative population dose at a favorable cost-benefit ratio, and the proposed gaseous waste treatment and ventilation systems meet the requirements of Sect II.D of Appendix I to 10 CFR Part 50.[1]

The staff concludes that the gaseous radwaste system for Units 2 and 3 is capable of maintaining releases of radioactive materials in gaseous effluents to "as low as is reasonably achievable" levels in accordance with 10 CFR Part 50.34a and meets the requirements of Appendix I to 10 CFR Part 50. The staff, therefore, concludes that the proposed system is acceptable.

3.2.3.3 Solid wastes

The solid waste system will be designed to process two general types of solid wastes: "wet" solid wastes which require solidification prior to shipment, and "dry" solid wastes which require packaging and, in some cases, compaction prior to shipment to a licensed burial facility. "Wet" solid wastes will consist mainly of spent filter cartridges, demineralizer resins, and evaporator bottoms which contain radioactive materials removed from liquid streams during processing. "Dry" solid wastes will consist mainly of low-activity ventilation air filters, contaminated clothing, paper, and miscellaneous items such as laboratory glassware and tools. Spent resins from the demineralizers will be collected in the spent resin storage tank. When the resin is to be packaged, it will be sluiced to a disposable liner and dewatered before solidification. The resin beads are solidified by filling the void spaces with urea formaldehyde and catalyst. A disposable paddle is used to agitate the mixture in the liner during the solidification process. Concentrated evaporator wastes will be collected in an evaporator bottoms tank, and then pumped batchwise through an inline mixer where they are blended with a urea formaldehyde solution. From the inline mixer, the mixture is sprayed into a disposal liner while a liquid catalyst is simultaneously sprayed into the liner by a separate nozzle to assure intimate mixing of the waste-urea formaldehyde solution and the catalyst.

On the basis of its evaluation and on recent data from operating plants, the staff has determined that about 425 m^3 (15,000 ft^3) per unit of "wet" solid wastes, containing about 1060 Ci of activity, will be shipped offsite annually. The principal radionuclides in the solid wastes will be long-lived fission and corrosion products, mainly Cs-134, Cs-137, Co-58, Co-60 and Fe-55. The applicant estimated the combined production of solid wastes from Units 2 and 3 to be 283 m^3/yr (10,000 ft^3/ year) of solidified wastes. The applicant calculated the total curie content of these solid wastes to be about 6500 Ci. The waste containers will be stored in a shielded area, as required, to reduce contact radiation levels.

Dry solid wastes will be packaged in cardboard boxes, wooden boxes, and special DOT-approved containers. Compressible wastes such as clothing and rags will be compressed prior to packaging. The staff estimates the dry solid wastes to total 283 m^3 (10,000 ft^3) per unit per year with a total activity content of less than 5 Ci. The applicant estimates the combined production of dry wastes from Units 2 and 3 to be 207 m^3/yr (7300 ft^3/year) with a calculated total curie content of about 21 Ci.

3.2.4 Chemical, sanitary, and other waste effluents

3.2.4.1 Chemical effluents

Several design changes have had significant impacts on chemical discharges. The condenser tubes are made of titanium (ER, Table 3.4-1) rather than of a copper-nickel alloy; this should eliminate the small amounts of copper and nickel in the discharge as described previously

(FES-CP, Sect. 3.5.1). An Amertap condenser tube cleaning system has been installed (ER, Sect. 3.4.4). In this system, sponge rubber balls are injected into the inlet piping of the condenser and are forced through the condenser tubes to scrape them clean. The balls are collected in the circulating water discharge conduit and are recirculated. This change helps to control fouling within the circulating water system and should reduce the frequency of chlorination necessary to maintain a clean condenser system. A makeup demineralizer system will replace the flash evaporators. Chemicals originally indicated as being discharged from the flash evaporators (FES-CP, Table 3.9) will not be discharged. A cellulose sealant for the circulating water system (FES-CP, Sect. 3.5.1) will not be used. Steam generator blowdown will be treated by filtration and demineralization and will be recycled to the condenser. Phosphates will not be added to the blowdown (FES-CP, Sect. 3.5.2), and the discharge of salts and heavy metal ions will be eliminated.

The only significant chemical discharge results from the use of sodium hypochlorite as a biocide. The chlorination system is common to both Units 2 and 3. The two units will not be treated at the same time. Hypochlorite solution will be injected into the circulating water pump discharge headers three times each day. Each injection will last about 15 min but will not exceed 90 min per unit per day. The chlorine residual in the circulating water discharge line is monitored by amperometric titration, and the addition of hypochlorite is adjusted to maintain a 0.5-mg/liter (1.89 grains/gal) maximum concentration of free available chlorine. The applicant estimates that this will result in a maximum free available chlorine concentration of 0.1 mg/liter (0.38 grains/gal) in the immediate vicinity of the discharge.

Other chemicals may be discharged at certain times. These chemicals generally will be discharged at low concentrations and, when mixed with the circulating water flow, represent a negligible concentration at the discharge to the ocean. During restarts the discharge of condensate from the hotwell may contain concentrations of several milligrams per liter of iron and copper. These substances will be reduced to negligible concentrations in the circulating water discharge. The discharge from the regeneration of demineralizers will contain sodium and sulfate ions; the concentrations at the discharge to the ocean will be less than 10 mg/liter (38 grains/gal) – negligible concentrations as compared to the natural concentrations in seawater. Small amounts of oil, not to exceed 5 mg/liter (19 grains/gal), will be discharged from the oil removal system and diluted to negligible concentration in the circulating water discharge. Various closed-loop cooling systems will be treated with potassium chromate to inhibit corrosion.

Offsite rainfall runoff from the coastal hills and from Interstate Highway 5 (I-5) is collected by the storm runoff drainage system for the highway. Part of this drainage is discharged directly to the ocean and part is discharged with the onsite plant drainage. Onsite plant drainage is collected in catch basins and is discharged with the circulating water discharge. Drainage collected in areas in which significant quantities of oil or grease might be present are routed through the oil removal system.

A National Pollutant Discharge Elimination System (NPDES) permit for SONGS 2 & 3 was issued on June 14, 1976, by the California Regional Water Quality Control Board, San Diego Region. The chemical effluent limitations for the combined discharges (cooling water, low-volume wastes, and storm drains) are: (1) the monthly average free available chlorine discharged shall not exceed 0.2 mg/liter (0.757 grains/gal), and the daily maximum shall not exceed 0.5 mg/liter (1.89 grains/gal); (2) discharge of free available chlorine or total residual chlorine from any plant unit for more than 2 hr in any one day or for more than one unit in the plant at any one time is prohibited; (3) the pH of the effluent shall be within the range of 6.0 to 9.0; and (4) after July 1, 1976, the discharge shall not exceed the limits given in Table 3.4. The permit prohibits the discharge of any chemicals or pollutants from the fish handling system. The low-volume waste discharge shall not exceed the following limits: (1) a monthly average of 30 mg/liter (113.6 grains/gal) and a daily maximum of 100 mg/liter (378.6 grains/gal) for total suspended solids and (2) a monthly average of 15 mg/liter (56.78 grains/gal) and a daily maximum of 20 mg/liter (75.7 grains/gal) for oil and grease. The discharge from the storm drains shall not exceed a monthly average of 10 mg/liter (38 grains/gal) and a daily maximum of 15 mg/liter (56.78 grains/gal) for oil and grease.

3.2.4.2 Sanitary and other waste effluents

Sanitary wastes from Units 2 and 3 will receive secondary level treatment in the sewage treatment plant located at Unit 1, which will serve all three units. The treated wastes will have the following water quality characteristics (average daily concentration): suspended solids, 30 mg/liter (113.6 grains/gal); biological oxygen demand, 30 mg/liter(413.6 grains/gal); coliform, mean probable number of 200 per 100 ml (59 per ounce); pH, 7.0 to 8.5; and total residual chlorine, 2.0 mg/liter (7.57 grains/gal) (ER, Table 5.4-1). The treated wastes will be discharged into the Unit 1 circulating water discharge at an average rate of about 0.02 m^3/min (5 gpm). Because the circulating water discharge at Unit 1 is about 1200 m^3/min (320,000 gpm), the sanitary waste effluents will be reduced to negligible concentrations at the point of discharge to the ocean. The sanitary waste effluents for all three units will be within the

Table 3.4. NPDES chemical effluent limitations

Constituent	Concentration (mg/liter) not to be exceeded more than	
	50% of time	10% of time
Arsenic	0.01	0.02
Cadmium	0.02	0.03
Total chromium	0.005	0.01
Copper	0.2	0.3
Lead	0.1	0.2
Mercury	0.001	0.002
Nickel	0.1	0.2
Silver	0.02	0.04
Zinc	0.3	0.5
Cyanide	0.1	0.2
Phenolic compounds	0.5	1.0
Total chlorine residual	1.0	2.0
Ammonia (as N)	40	60
Total identifiable chlorinated hydrocarbons	0.002	0.004
Toxicity concentration	1.5[a]	2.0[a]

[a]Toxicity units.
Source: ER, Appendix 12C.

(To convert mg/liter to grains/gal, multiply by 3.785.)

limitations established for Unit 1 by the California Regional Water Quality Board and the Environmental Protection Agency.

Some gaseous wastes from the operation of diesel generators and the auxiliary boiler will be discharged intermittently. Four diesel generators will serve Units 2 and 3, and it is anticipated that these will operate for about 2 hr once per month. The estimated hourly full-load emission in kilograms (pounds) from each generator is nitrogen oxides, 84 (185); sulfur dioxide, 11 (25); particulates, 0.9 (2); hydrocarbons, 3.9 (8.5); and carbon monoxide, 9.5 (21) (ER, Sect. 3.7.4.1). A single auxiliary boiler will be used for both Units 2 and 3. This boiler will be operated for varying time periods throughout the life of the plant (ER, Sect. 3.7.4.2). The maximum annual use is expected to be 1250 hr at full load and 3130 hr at half load. Under these conditions, the anticipated annual emissions in tonnes (tons) are nitrogen oxides, 44 (49); sulfur dioxide, 98 (108); and particulates, 34 (38).

Trash from screens for the circulating water system for Units 2 and 3 will be taken to the Bonsall Sanitary Landfill near the city of Vista, California. This landfill is used for the disposal of trash from Unit 1.

3.2.5 Transmission lines

Much of the description of the transmission lines presented in Sect. 3.7 of the FES-CP is no longer valid. Construction of SCE's transmission line from SONGS to Santiago Substation will be completed only up to Santiago Tap, thereby deleting that portion between Santiago Tap and Santiago Substation. SDG&E's line from Telega Substation to Escondido Substation has also been deleted. SCE will retrofit transmission lines from SONGS to Santiago Tap, Santiago Tap to Santiago Substation, and Santiago Tap to Black Star Canyon Tap. SDG&E will add a line from SONGS to Mission Substation. SDG&E's lines from SONGS to Telega Substation and SONGS to Encina Substation will still be constructed but the staff has received additional information with regard to these lines since issuance of the FES-CP. Therefore, these lines will be further discussed in Sect. 3.2.5.2. All transmission lines for operation of SONGS Units 2 and 3 are illustrated in Figs. 3.7 and 3.8. Generally, the lines are coastal, using existing rights-of-way traversing northward from SONGS to Talega Substation, Santiago Tap, Santiago Substation, and Black Star Canyon Tap, and southeast to Encina and Mission Substations. A total of about 159.1 km (98.9 miles) will be crossed by the transmission lines. No new rights-of-way, however, will be required.

The SCE and SDG&E transmission lines will each be supported by two steel horizontal portal structures (Fig. 3.9) for the initial 0.6 km (0.4 mile) of right-of-way northeast of the SONGS switchyard. These structures will replace the steel lattice towers now supporting the existing circuits in this area. No additional land for tower bases or access roads will be required.

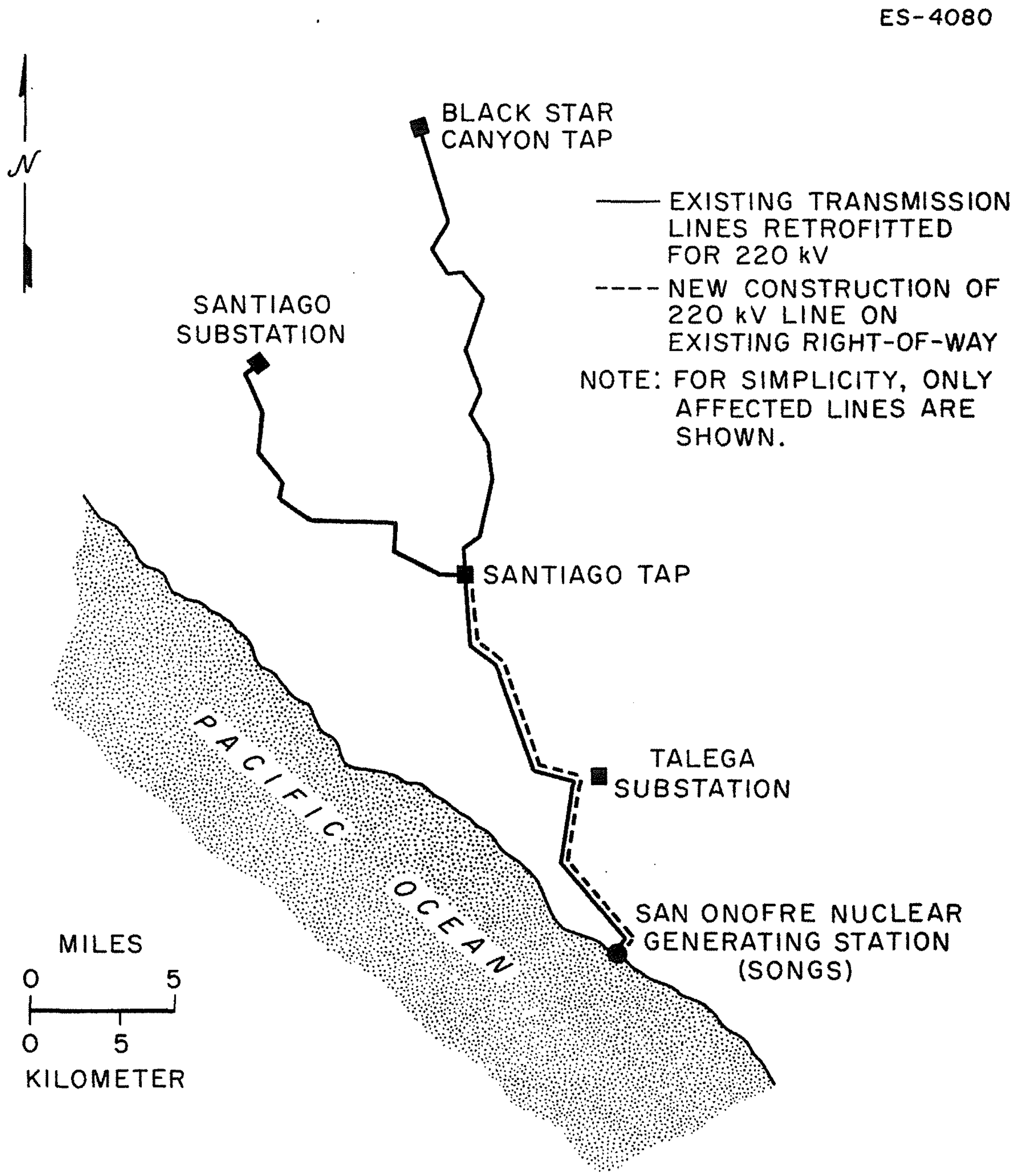


Fig. 3.7. *Schematic diagram of proposed Southern California Edison Company* transmission lines for SONGS 2 & 3.

3.2.5.1 SCE transmission lines

A double circuit 220-kV transmission line will be constructed between SONGS and Santiago Tap, an approximate distance of 24.3 km (15.1 miles) (Fig. 3.7). About 73 steel lattice towers (Fig. 3.10) will be required for this line, with an average span of about 335 m (1100 ft) between towers. The average tower height is estimated to be 39.6 m (130 ft). The new tower bases will require 2.44 ha (6.03 acres), and access road extensions are expected to require 1.32 ha (3.25 acres) of land (ER, Suppl. 2, Item 36). Additional transmission lines required by SCE that were not discussed in the FES-CP are those from SONGS to Santiago Tap, Santiago Tap to Santiago Substation, and Santiago Tap to Black Star Canyon Tap. These lines, totaling

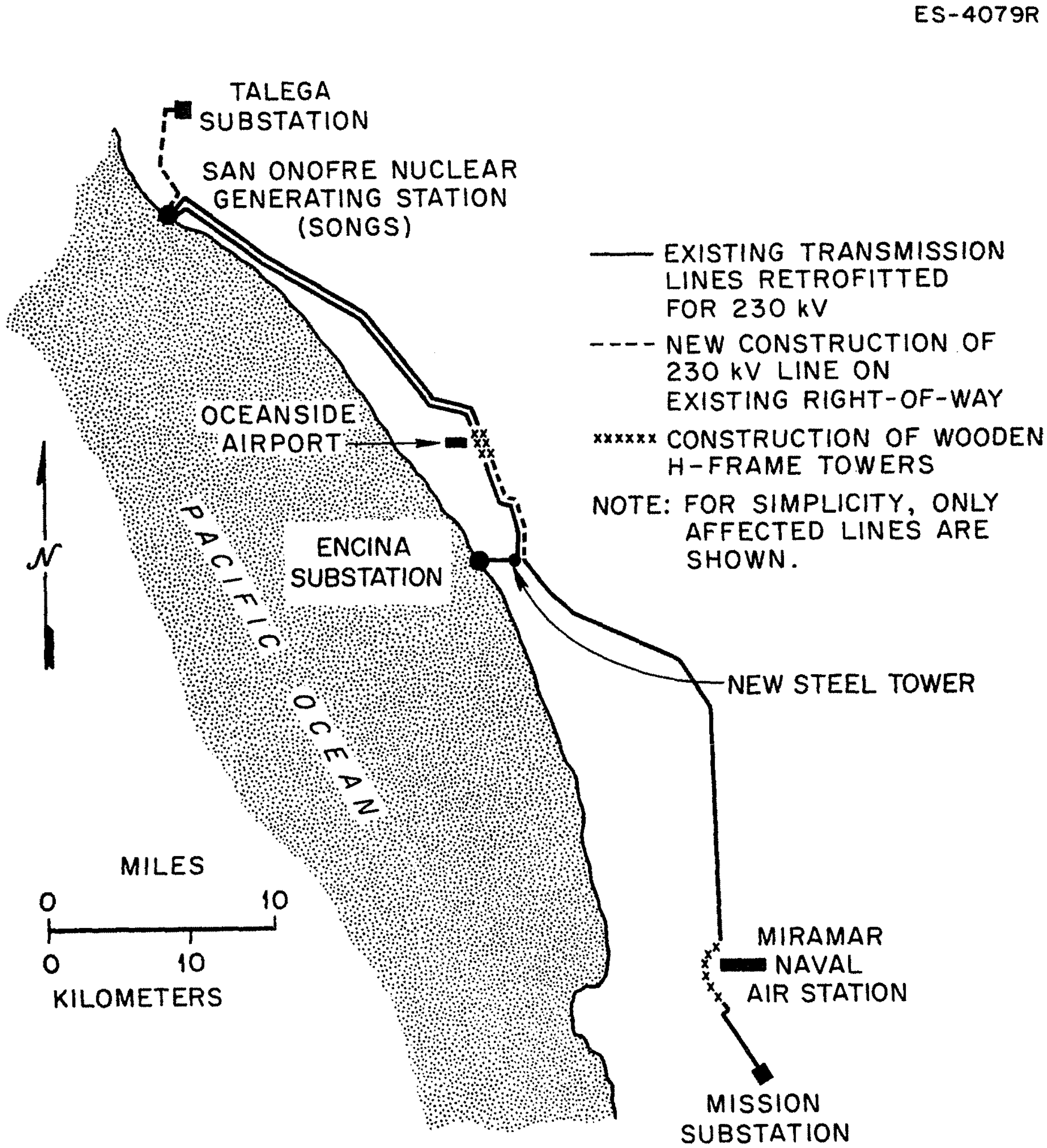
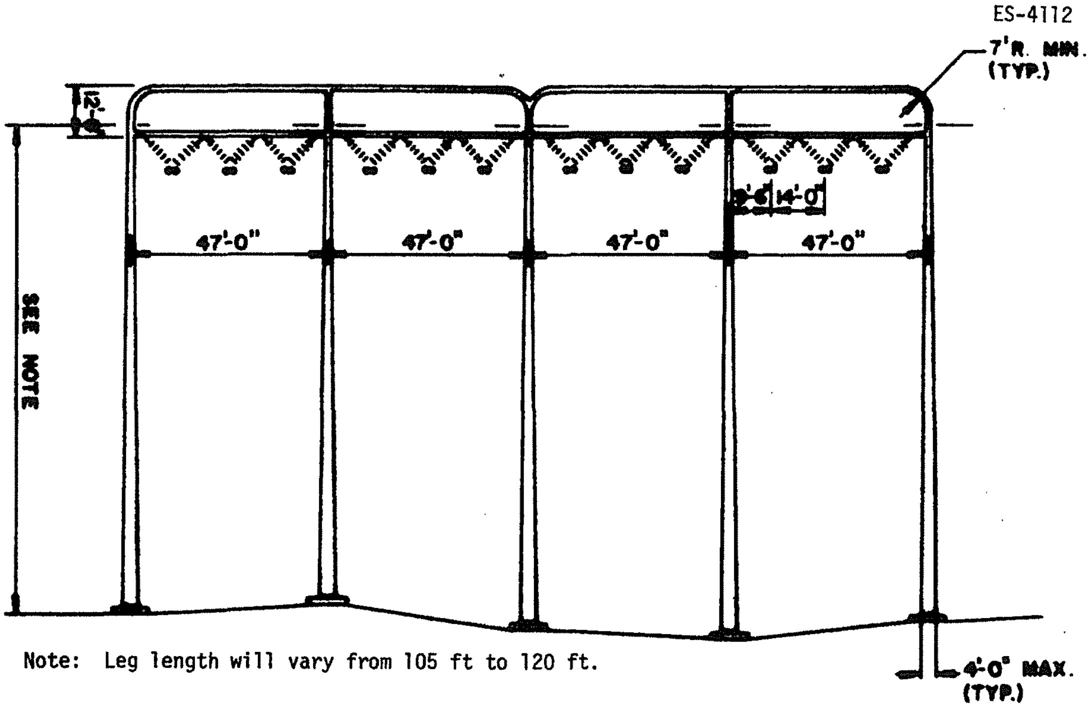


Fig. 3.8. Schematic diagram of proposed San Diego Gas and Electric Company transmission lines for SONGS 2 & 3.

71.7 km (44.2 miles) will be retrofitted to operate at 220 kV. Retrofitting will involve the replacement of existing conductors with larger ones (on existing towers) and the construction of four additional towers between Santiago Tap and Black Star Canyon Tap.[4] These towers are required to provide adequate ground clearance in some spans where the wire tension will have to be reduced from its present value (ER, Sect. 3.9.1.1). This additional construction is expected to require 0.13 ha (0.33 acres) of land for new tower bases and 0.52 ha (1.3 acres) for access road extensions (ER, Suppl. 2, Item 36).

The material storage yard for SCE transmission lines will be located about 1.6 km (1 mile) north of the San Onofre Nuclear Generating Station within Camp Pendleton Marine Base. The area involved will be about 2.2 ha (5.5 acres) and will not require any clearing or opening of new roads (ER, Suppl. 2, Item 30).



Note: Leg length will vary from 105 ft to 120 ft.

Fig. 3.9. Four-circuit steel horizontal portal structures used by Southern California Edison Company; San Diego Gas and Electric Company will use a similar structure with five circuits. Source: ER, Fig. 3.9-2.
(To convert ft to m, multiply by 0.3048.)

3.2.5.2 SDG&E transmission lines

The only transmission line required by SDG&E that was not discussed in the FES-CP will run between SONGS and Mission Substation, a distance of 85 km (53 miles) (Fig. 3.8). This line will be installed by adding a 230 kV circuit to the vacant position on existing double circuit towers;[1] some of the existing towers will be replaced. A total of about 36 wooden H-frame towers (Fig. 3.11) will be constructed along a 1.6-km (1-mile) segment east of Oceanside Airport and a 6.8-km (4.2-mile) segment opposite Miramar Naval Air Station to accommodate FAA regulations.[1] About 9 km (5.6 miles) of existing 138 kV wood structures south of the Oceanside Airport will be replaced by approximately 32 double circuit steel lattice towers (Fig. 3.12). The construction of the new towers for this line will not require any additional land for tower bases or access roads (ER, Suppl. 2, Item 36). Subsequent to issuance to the FES-CP, additional information was supplied by the applicant regarding the line from SONGS to Encina Substation and SONGS to Talega Substation. The line from SONGS to Encina Substation, 40 km (25 miles), will be formed by adding a 230 kV circuit to the vacant position on existing double circuit towers.[1] In addition, approximately four wooden H-frame towers (Fig. 3.11) will be constructed along a 1-km (0.6 mile) segment east of Oceanside Airport to accommodate FAA regulations. To facilitate arrangement of the new conductors, a single steel tower will also be installed east of Encina Substation. All new structures will be constructed within existing rights-of-way and will not require any additional land for tower bases or access roads (ER, Suppl. 2, Item 36). The line from SONGS to Talega Substation traverses about 11.3 km (7 miles) and will require construction of about 32 steel lattice towers (Fig. 3.12). The new tower bases will require about 0.23 ha (0.58 acre), and access road extensions are expected to require 0.53 ha (1.3 acres) of land (ER, Suppl. 2, Item 36). Because SDG&E's original plan assumed that the Talega Substation would be constructed and in operation prior to completion of SONGS 2 & 3 (ER, Suppl. 2, Item 25), this facility was discussed in the FES-CP as if it were already in existence. Construction, however, was delayed. The proposed Talega Substation is expected to cover 2 ha (5 acres) of land; an additional 2 ha (5 acres) around the substation will also require grading.

The material storage yard for SDG&E transmission lines will be located in existing substations with the following exceptions: (1) fencing a level area of about 0.09 ha (0.23 acre) adjacent

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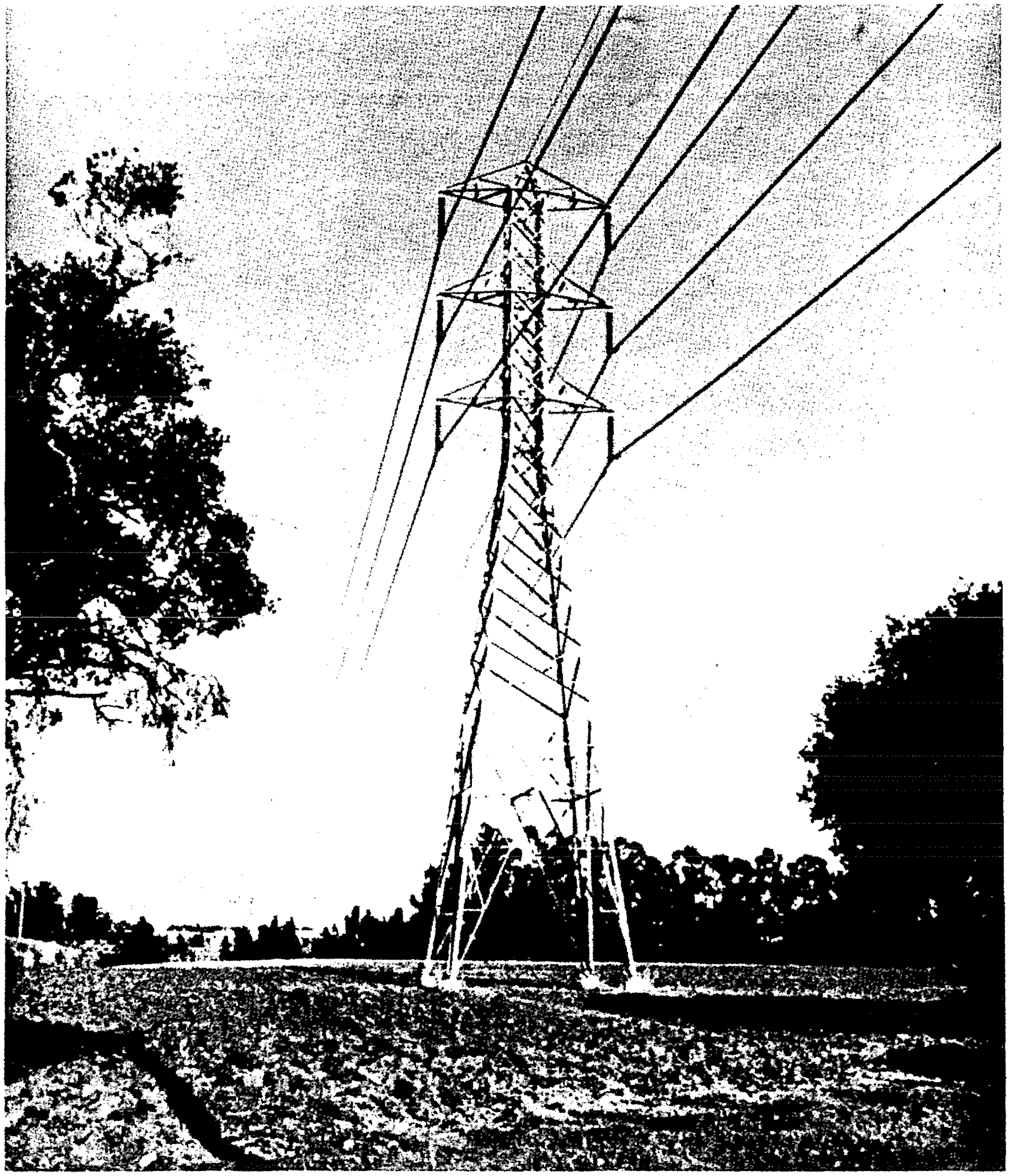

Fig. 3.10. Typical steel lattice tower design used by Southern California Edison Company. Source: ER, Fig. 3.9-3.

to the existing Pulgas Substation and (2) fencing a level area of about 0.09 ha (0.23 acre) adjacent to the Japanese Mesa Substation. No grading, clearing, or additional access roads are anticipated for this project (ER, Suppl. 2, Item 30).

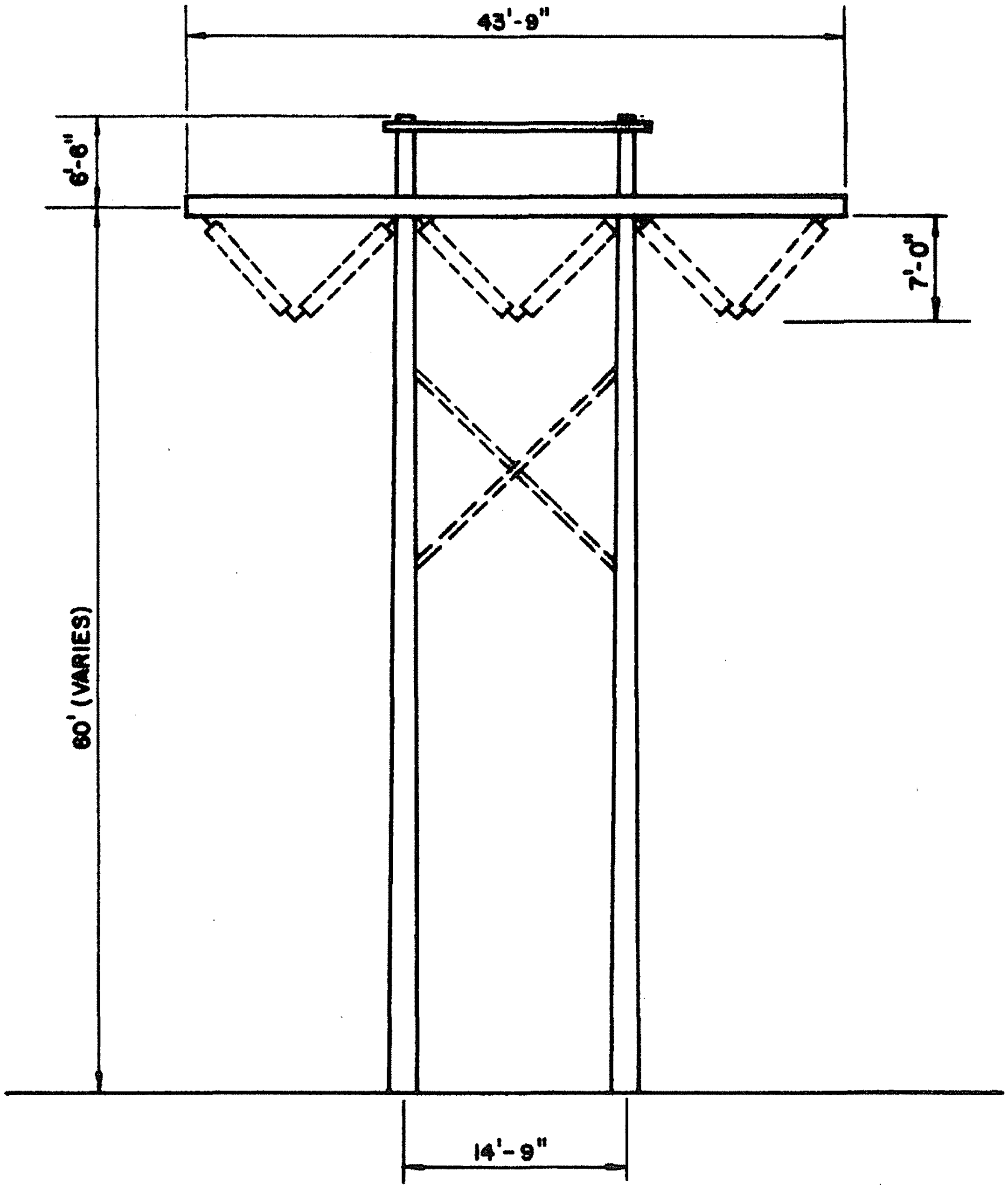

Fig. 3.11. Wooden H-frame tower used by San Diego Gas and Electric Company. Source: ER, Fig. 3.9-9.(To convert ft to m, multiply by 0.3048; to convert in. to mm, multiply by 25.4.)

3.2.6 Probable maximum flood berm

3.2.6.1 Description of structure and existing environment

Subsequent to issuance of the FES-CP the applicant was required to construct an earthern berm to protect the Station form the probable maximum flood (PMF). Construction of this structure

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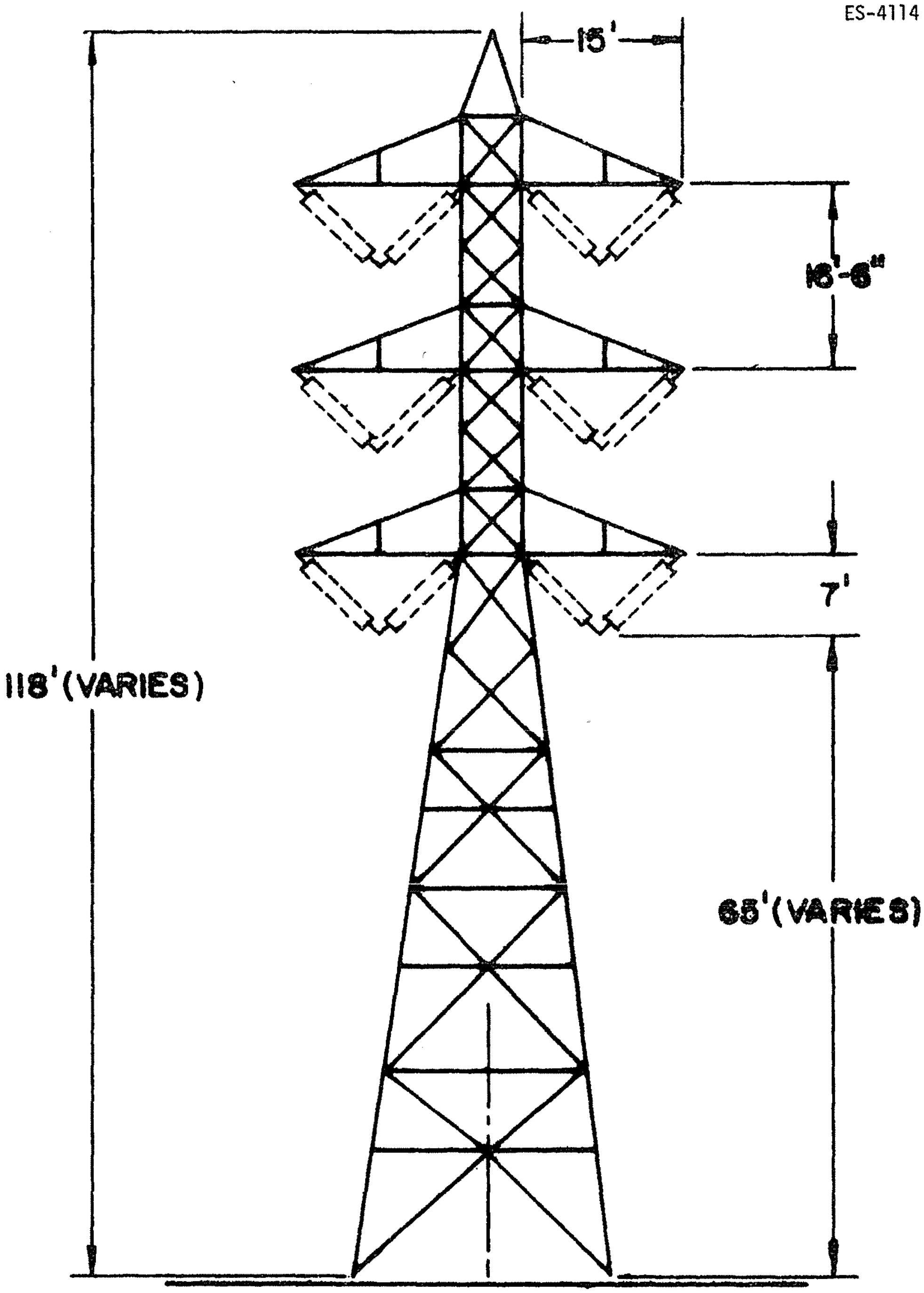

Fig. 3.12. Typical steel lattice tower design used by San Diego Gas and Electric Company. Source: ER, Fig. 3.9-8.(To change ft to m, multiply by 0.3048.)

and associated environmental impacts are presented by the applicant in a letter to the NRC[5] and in the applicant's final safety analysis report (FSAR).

The San Onofre site is located on a coastal plain at the base of the western foothills of the Santa Margarita Mountain Range. Elevation in this area rises sharply from sea level to a fairly level terrace formation 30 to 61 m (100 to 200 feet) above sea level. About 450 m (1500 feet) inland the foothills begin, rising with moderate slopes to an elevation of about 900 m (3000 ft) above sea level. Natural plant cover in the coastal plain typically consists of coastal chaparral and grassland, while in the foothills it is composed primarily of chaparral and open woodland.

There are no perennial streams in the general vicinity of the plant site. However, ephemeral streams and water courses do exist. The major streams are San Mateo Creek, located about 3.2 km (2 miles) to the northwest and San Onofre Creek located approximately 1.6 km (1 mile) to the northwest. The drainage divide separating San Mateo and San Onofre Creeks precludes the plant site from being influenced by San Mateo Creek. The applicant's results of the probable maximum flood (PMF) analysis concluded that the San Onofre Creek Basin exhibits no flooding potential to the site (FSAR, Sect. 2.4.2.2). Topographical features of the basin would contain the maximum flood stage and thereby preclude flooding of the site by this source. The foothill drainage basin, however, does contribute to the hydrologic factors influencing the plant site. The basin totals 2.2 km^2 (0.86 mi^2). There are no gaging stations located within the basin and, consequently, stream flow records are not available.

The entire watershed of the foothill drainage basin lies within the boundaries of the Marine Corps Base, Camp Pendleton. Elevation of the basin varies between 30 to 365 m (100 to 1200 feet) above sea level. Ground slope varies from 8 to 22%. Ground cover is moderate, consisting mainly of chaparral and grassland.

Water control structures at the foot of this basin consist of the 107- and 183-cm (42- and 72-in.) diameter concrete culverts under I-5. The capacity of these culverts is 5.1 and 14.7 m^3/sec (180 and 520 ft^3/sec), respectively. In addition to the two culverts, an earthern channel traverses the basin along the east side of I-5 diverting runoff to San Onofre Creek. The capacity of the channel is 52.4 m^3/sec (1850 ft^3/sec).

The applicant's analysis of the flooding potential of the foothill drainage area indicated that the plant site could be subjected to flooding during the occurrence of the PMF. In order to preclude flooding of the site by this source a diversion structure routes the surface runoff from the foothill drainage area to the San Onofre Creek Basin. This PMF structure will be an earthern berm, having an isoceles trapezoid cross section that is 2.4 m (8 feet) high and 12.8 m (42 feet) wide at its base, with 2:1 side slopes. The berm will parallel I-5 and will be 2.7 km (1.68 miles) long. The existing channel which parallels the proposed berm will be widened where necessary and will vary from 7.6 to 30.5 m (25 to 100 ft) in width. The berm will cover a portion of an existing road, El Camino Real Road, requiring the construction of a new road. The relocated road will run approximately parallel to and east of the proposed PMF berm.

Relocation of the road will require about 1.4 ha (3.5 acres) of land, the berm will cover approximately 3.5 ha (8.6 acres), and the channel (assuming a 30 m (96 ft) width) will require about 8.3 ha (20.6 acres) for a total land area requirement of 13.2 ha (32.7 acres). The existing channel and El Camino Real Road are included in this acreage.

A terrestrial biological survey of the site was conducted on October 25 and 31, 1977. Vegetation on the site is basically a southern coastal sage scrub community, being influenced by the coastal marine climatic conditions. However, nearly half of the site (northern portion) has been previously disturbed as evidenced by the presence of many non-native "weedy" species including saltbush (Atriplex semibaccata), Russian thistle (Salsola kali), mustard (Brassica geniculata) tree tobacco (Nicotiana glauca), and sow thistle (Sonchus oleraceus). Native species on this area include California sagebrush (Artemesia californica), California buckwheat (Eriogonum fascilculatum), and coyote brush (Baccharis pilularis). The southern half of the site is primarily vegetated with native species of the coastal sage scrub plant community including the native species listed above. The land on which the El Camino Real Road will be relocated contains many of the same species that occur at the berm site, but with a higher degree of cover.

Fauna surveys of the site and vicinity demonstrated that the majority of the species present were birds (24 species). Red-tailed hawks (Buteo jamaicensis) were prevalent in the vicinity using wooden posts, telephone and power poles as perches and a SCE lattice transmission tower for nesting. Although only 2 species of reptiles and 2 species of mammals were observed, others are likely to occur in the vicinity.

No threatened or endangered flora or fauna were observed on the proposed PMF Berm site, the area to be cut, or on the area where the El Camino Real Road is to be relocated.[5]

On November 14, 1977, an onsite inspection of the alignments of both the proposed berm and access road was conducted to determine the presence or absence of surficial paleontologic

values.[5] Although the survey did not result in locating any fossils, a review of the literature revealed that all sedimentary formations in the vicinity contain fossils. No localities in the immediate area have been placed on the National Registry of Natural Landmarks.

The site was surveyed for archaeological resources on December 8, 15, and 16, 1977 (ref. 5). The northern third of the berm was not surveyed because it had previously been studied; some portions of the berm also were not adequately surveyed because of dense vegetation.[5] In one area, eight pieces of marine shell were observed. The shells, however, were weathered and worn and gave the appearance of paleontological specimens, rather than archaeological remains.[5] An archaeological map and literature search revealed four recorded archaeological sites within 1.6 km (1 mile) of the proposed project, but none were located within the project area.[5]

3.2.6.2 Impacts of PMF berm

The berm will be built on top of an existing asphalt road. Consequently disruption of this area will have no significant biological impact. Widening the existing channel which parallels the proposed berm will require loss of about 8.5 ha (21 acres), and an additional 1.4 ha (3.5 acres) of habitat will be lost due to relocation of El Camino Real Road. Because these habitats do not represent unique communities, loss of this relatively small acreage should have no significant impact to biological resources of the area. To minimize the impact to raptors nesting in the vicinity the applicants will attempt to avoid construction activity during the period of March and April.[5]

The construction of the PMF berm might physically destroy fossils and/or relationships between fossils, or the environmental context of original deposition, that could provide significant paleontological data. In addition, the berm and new road may cover deposits containing significant paleontological data thereby making such data unreachable. To mitigate these potential impacts the applicants will conduct a paleontological survey prior to construction and monitor the excavation as it proceeds.[5] This will allow fossils to be salvaged as they are unearthed. Construction should be phased so that equipment could be shifted to other areas if fossils were located. Sufficient time should be allowed to uncover, record, and remove the fossils. If excavation were initiated in areas of highest paleontological potential, equipment could be moved to areas of low potential if paleontological values were encountered. This would provide a maximum amount of construction time and a maximum amount of time for paleontologic resource recovery.

Construction of the proposed PMF berm should not cause any direct or indirect adverse impact to known archaeological resources. However, the site would have been a favorable area for aboriginal habitation; i.e., an area of relatively flat topography with abundant fresh water and food resources.[5] The probability exists that buried resources may be in the area, especially where dense vegetation obscures the surface. Consequently, a trained archaeologist will monitor the construction activity and take appropriate conservation measures if necessary.[5]

No significant commitments of resources will result from construction and maintenance of the PMF Berm. The possibility exists that potential archaeological or palcontological resources would be destroyed during the excavation activity required for construction of the berm. However, if the proper mitigation measures are performed (monitoring, analysing, interpretating, preserving, and reporting), then these resources would not be irretrievable.

3.2.6.3 Floodplain management

The objective of Executive Order 11988, "Floodplain Management," is ". . . to avoid to the extent possible, the long- and short-term adverse impacts associated with the occupancy and modification of floodplains and to avoid direct and indirect support of floodplain development whenever there is a practicable alternative." The Construction Permit was issued and the majority of construction completed prior to issuance of the Executive Order. Thus we conclude that no practicable alternative locations exist. The following is a discussion of floodplain conditions prior to construction of the plant and alterations made to these floodplains as a result of construction of San Onofre Units 2 and 3.

The San Onofre Units 2 and 3 are bounded on the east by Interstate Highway 5, the Atchison Topeka and Santa Fe Railroad and Highway 101. Interstate Highway 5 was constructed in 1968 prior to San Onofre Units 2 and 3. As part of the I-5 construction, a drainage channel designed for 100-year storm runoff was constructed parallel to and east of I-5. This channel intercepted tributary rainfall runoff from the foothills east of I-5 and transported it to the north away from the plant. The channel then merged with San Onofre Creek which in turn flowed to the Pacific Ocean.

The plant site which is bounded on the west by the Pacific Ocean was originally on a high coastal bench approximately 100 feet above sea level. Located at this elevation, the site was protected from severe flooding events and thus was not in the 100-year ocean floodplain.

The existing drainage channel which is west of and parallel to I-5, is being enlarged to contain floods and debris. The design capacity of the channel enlargement and extension is the Probable Maximum Flood, an event which is greater than the one-percent chance flood. The improvement will not induce higher flood stages.

The San Onofre plant grade is lower than the original coastal bench. However, construction of a seawall on the seaward side of the plant and east of the original bluff line provides protection from events larger than the one percent chance flood.

The plant, including the intake structure and seawall, is not built in the 100-year floodplain and will not be flooded by any 100-year flood levels. The intake crib and intake and discharge conduits are submerged on the ocean floor. The channel improvement east of Interstate Highway 5 will not increase flood levels. Therefore, the construction and operation of the San Onofre Unit 2 and 3 Nuclear Generating Station will comply with the intent of Executive Order 11988.

3.2.7 Emergency facilities

Emergency plans for San Onofre Units 2 and 3 call for an onsite Technical Support Center adjacent to the control room and an interim onsite Operational Support Center in the lunch room of the administration, warehouse, and shop building. Neither requires changes in the structural design or layout of the facility. An offsite Emergency Operations Facility is tentatively planned to be constructed on Japanese Mesa, east of Interstate 5, within the Camp Pendleton Reservation. This area was used for disposal of excavated material during construction. The structures must be designed to accommodate a minimum of 35 people.

REFERENCES

1. U.S. Nuclear Regulatory Commission, "Numerical Guides for Design Objectives and Limiting Conditions for Operation to Meet the Criterion 'As Low as Practicable' for Radioactive Material in Light-Water-Cooled Nuclear Power Reactor Effluents," May 5, 1975, and as amended Sept. 4, 1975, and Dec. 17, 1975, in Title 10, "Code of Federal Regulations," Part 50, Appendix I.

2. U.S. Nuclear Regulatory Commission, "Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Pressurized Water Reactors (PWR-GALE Code)," NUREG-0017, April 1976.**

3. U.S. Nuclear Regulatory Commission, "Cost-Benefit Analysis for Radwaste Systems for Light-Water-Cooled Nuclear Power Reactors," in Regulatory Guide 1.110, March 1976.**

4. Letter from Ira Thierer, Southern California Edison Co., to Dr. Knox Mellon, State Historic Preservation Officer, June 2, 1978.*

5. Letter from J. G. Haynes, Southern California Edison Co., to W. H. Regan, Jr., USNRC, undated, docketed on February 18, 1978.*

*Available for inspection and copying for a fee in the NRC Public Document Room, 1717 H St., N.W. Washington, DC 20555.

**Available from the NRC/GPO Sales Programs, Washington, DC 20555 and from the National Technical Information Service, Springfield, VA 22161.

4. STATUS OF SITE PREPARATION AND CONSTRUCTION

4.1 RESUME AND STATUS OF CONSTRUCTION

As of December 1980, the construction of SONGS Unit 2 was 97% complete, and SONGS Unit 3 was 68% complete. Figure 4.1 is a recent photograph of the site.

Impacts of construction have been identified in the FES-CP. The major terrestrial impact has been the excavation of about 16.4 ha (40.5 acres) of the San Onofre Bluffs, which resulted in the loss of a small amount of wildlife habitat. No rare or endangered animal species in the vicinity of the site have been or are expected to be adversely affected by construction activities.

The environmental impacts associated with changes in the routing of transmission lines subsequent to issuance of the FES-CP have been evaluated by the staff in its environmental impact appraisal regarding extension of the earliest and latest construction completion dates.

4.2 Offsite Emergency Operations Facility

An offsite Emergency Operations Facility is tentatively planned to be constructed on Japanese Mesa, east of Interstate 5, within the Camp Pendleton Reservation. This area was used for disposal of excavated material during construction. The structure must be designed to accommodate a minimum of 35 people. Construction of the Emergency Operations Facility on Japanese Mesa will not significantly disturb the area relative to previous disturbances.

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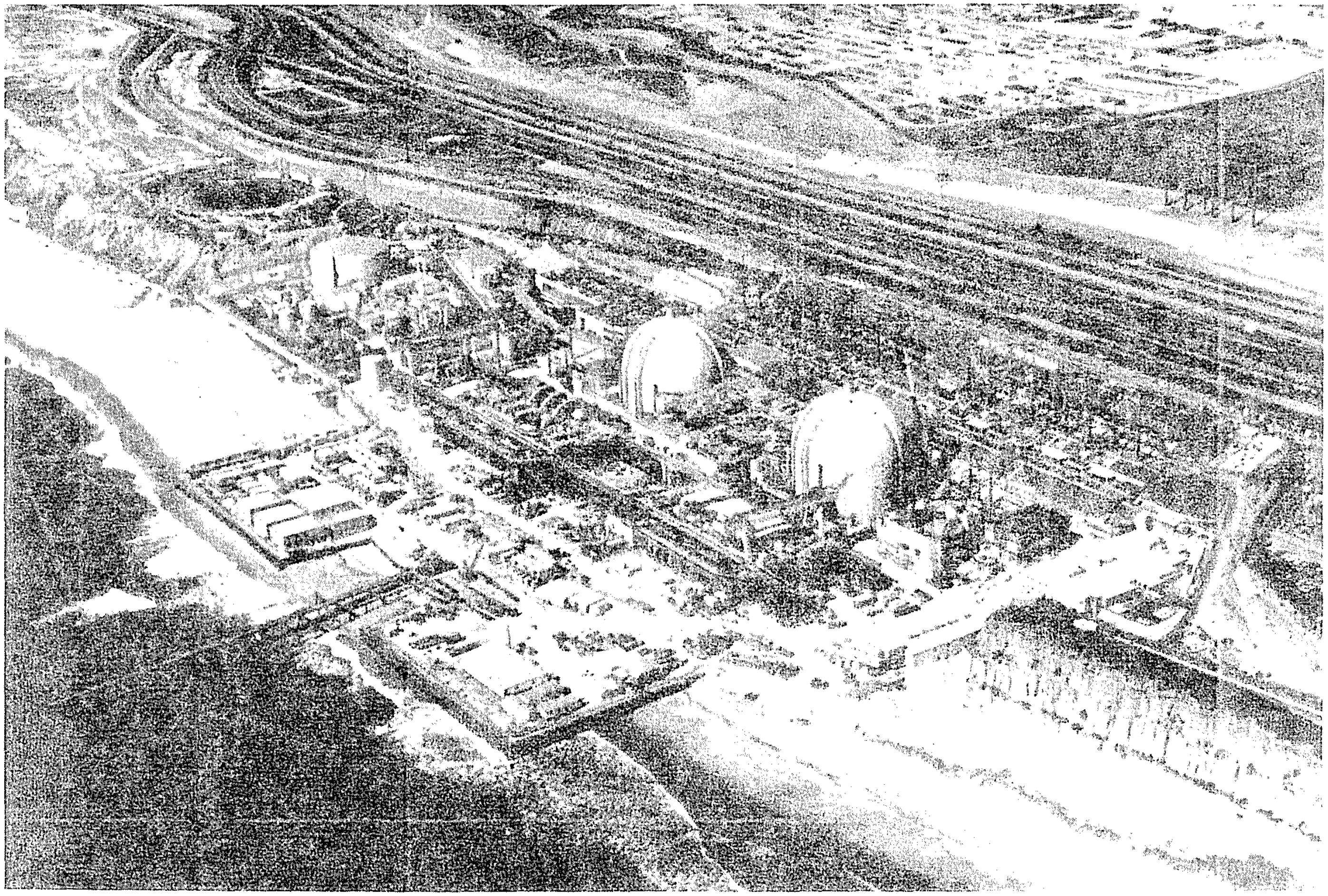

Fig. 4.1. Photograph of San Onofre Nuclear Generating Station taken in October 1980.

5. ENVIRONMENTAL EFFECTS OF STATION OPERATION

5.1 RESUME

The major design changes that have environmental effects involve the heat dissipation system. A more thorough analysis by the staff of the thermal plume is described in Sect. 5.3.1.2. The effects of the revised thermal-plume analysis on aquatic biota are discussed in Sect. 5.4.2.1. Changes in the effects of chemical effluents are discussed in Sects. 5.3.2 and 5.4.2.2. A revised discussion of radiological impacts is given in Sect. 5.5. Sect. 5.6 contains a revised assessment of the socioeconomic impacts.

5.2 IMPACTS ON LAND USE

Although the transmission line routes have been modified since the issuance of the construction permit (Sect. 3.2.5), the analysis of projected impacts as set forth in the FES-CP (Sect. 5.1) remains valid. All new transmission lines will be constructed on existing rights-of-way; a total of 5.2 ha (12.8 acres) of land will be required for access road extensions and for new tower bases.

The operation of SONGS 2 & 3 is not expected to affect any existing or proposed areas of the National Park System nor any existing or known potential sites to be listed as national landmarks.[1] In 1980, the applicant conducted a National Register assessment program of the 230 kV transmission right-of-way from San Onofre Nuclear Station to Black Star Canyon and Santiago Substation and to Encina and Mission Valley Substation and evaluated 41 previously identified archaeological sites. As a result of this effort, the NRC, in consultation with the State Historic Preservation officer, is seeking a determination of eligibility for inclusion in the National Register of Historic Places for 23 sites (see Appendix D, letter from Dr. Knox Mellon, State Historic Preservation officer, to D. C. Scaletti, USNRC, dated December 18, 1980). The staff agrees with the conclusions of the December 18, 1980, letter and will seek concurrence of determinations of effect from the Advisory Council on Historic Preservation.

5.3 IMPACTS ON WATER USE

5.3.1 Thermal discharges

5.3.1.1 Applicant's thermal analysis

The applicant retained the California Institute of Technology to perform a thermal analysis for the purpose of modifying the diffuser design in order to ensure compliance with state thermal standards. To accomplish this, a physical hydraulic model study was carried out at the W. M. Keck Laboratory of Hydraulics and Water Resources. The culmination of this effort was the diffuser design and configuration described in Section 3.2.2.

The physical model simulation was performed in a basin having horizontal dimensions of 11 m (36 ft) by 6 m (20 ft) which represents a prototype modeled region of about 8500 m (28,000 ft) by 4900 m (16,000 ft). The location and orientation of the Units 2 and 3 model intakes and diffusers within the basin are illustrated in Fig. 5.1. The bottom of the basin was filled with sand which was shaped to produce a simplified representation of the San Onofre bathymetry. The resulting bottom geometry was uniform in the longshore direction and varied as a composite of linear slopes in the offshore direction, as shown in Fig. 5.2. In order to satisfy scaling laws, the number of ports per laboratory diffuser was 16.

To perform simulations, the basin was filled with water at a constant temperature, then water at a temperature 16.67°C (30°F) higher was discharged through the diffusers. This excess temperature was required to maintain proper similitude and represents a 11.1°C (20°F) prototype excess temperature. Water was withdrawn from the basin through the intakes; however, this water was not recirculated. The model basin had the capability to simulate a variety of longshore current regimes, and among those investigated were no crossflow, crossflows of various amplitudes, reversing flows of various amplitudes, and special currents. The results of the simulations are summarized in the ER, Table 5.1-1. Among

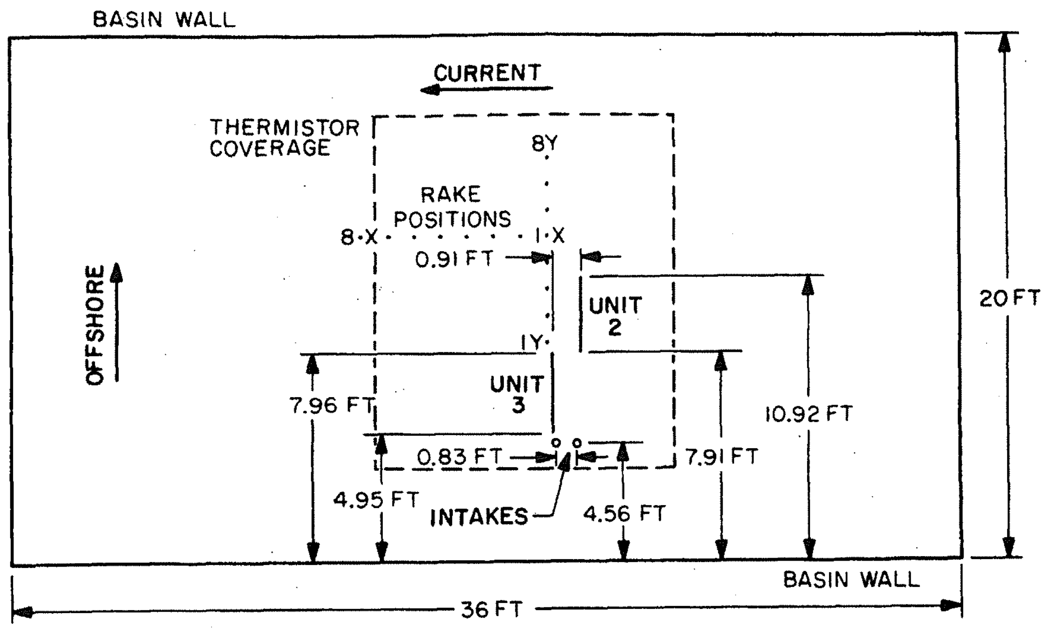

Fig. 5.1. Layout of basin used for the physical model study. Source: R. C. Y. Koh, N. H. Brooks, E. J. List, and E. J. Wolanski, *Hydraulic Modeling of Thermal Outfall Diffusers for the San Onofre Nuclear Power Plant*, W. M. Keck Laboratory of Hydraulics and Water Resources, California Institute of Technology, Report KH-R-30, January 1974, Fig. 6.1. (To change ft to m, multiply by 0.3048.)

these simulations, the worst case was that of zero crossflow. A plot of surface isotherms produced by the model for this case is given in Fig. 5.3. Further details of the physical-model study can be found in ref. 2. There are, however, certain physical conditions and mechanisms that could not be properly modeled in the laboratory. In an effort to account for this limitation on modeling, the modelers associated a probable temperature excess with each uncertainty. The total of these individual uncertainties was 0.83°C (1.5°F). It was therefore reasoned that state thermal standards should be met if the laboratory results satisfied these standards for 1.39°C (2.5°F), with the 0.83°C (1.5°F) margin of error, rather than 2.2°C (4.0°F).

It is evident from Fig. 5.3 that this case satisfies the state thermal standards. The applicant suggests that this is the worst case and, therefore, concludes that SONGS 2 and 3 will, under all conditions, comply with California State thermal standards.

The staff has reviewed the applicant's thermal analysis and believes that the physical model does not adequately represent certain hydrodynamic mechanisms and certain physical features of the prototype. The most significant of these is the duration of the physical model simulation. The staff believes that the physical model simulation, which yielded the result given in Fig. 5.3, has not reached thermal equilibrium. This is apparent in the applicant's results for surface excess temperature versus time given in Fig. 5.4. The upper curve represents the maximum surface temperature as a function of time anywhere in the basin, while the lower curve represents the maximum surface temperature as a function of time beyond 305 m (1000 ft) from the discharge point. The time scale for thermal equilibrium in the upper curve is a function of the time required for the heated water from the discharge to reach the surface and, therefore, should be relatively short. The staff has substantiated this by performing a least-squares curve fit on the data shown in the upper curve. The results show that the maximum surface excess temperature anywhere in the basin is increasing less than 0.028°C (0.05°F) per day. This is small compared with the standard deviation of the curve fit and, therefore, thermal equilibrium can justifiably be assumed. Beyond 305 m (1000 ft) from the discharge, the thermal equilibrium time scale will be a function of the rate of transport of heated water by densimetric effects and diffuser momentum away from the discharge point. This time scale should be longer than that for thermal equilibrium near the discharge. A similar curve fit performed on the lower plot reveals that the excess surface temperature beyond 305 m (1000 ft) from the discharge is increasing by approximately 0.16°C (0.29°F) per day. The staff believes that such a time-rate-of-change of temperature does not represent thermal equilibrium. Using a mathematic model, the staff has qualitatively reproduced the applicant's results. However, this mathematical simulation demonstrates that for increased

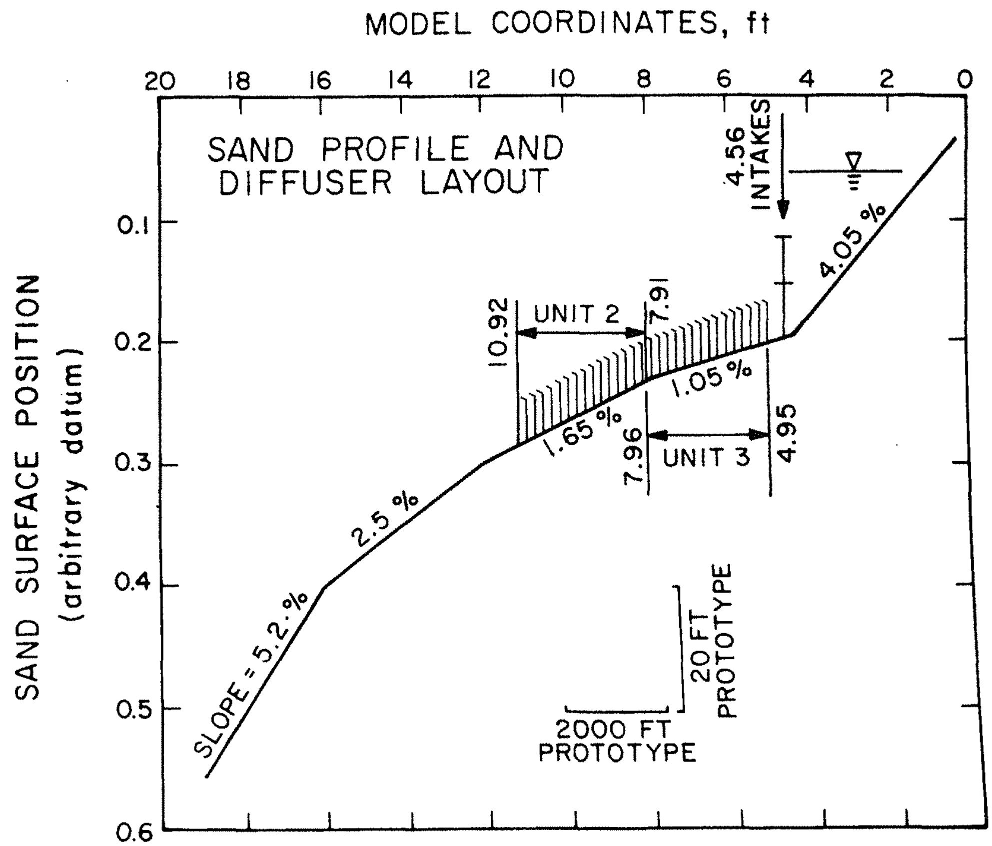

Fig. 5.2. Bottom profile used for the physical model study. Source: R. C. Y. Koh, N. H. Brooks, E. J. List, and E. J. Wolanski, "Hydraulic Modeling of Thermal Outfall Diffusers for the San Onofre Nuclear Power Plant," W. M. Keck Laboratory of Hydraulics and Water Resources, California Institute of Technology, Report KH-R-30, January 1974, Fig. 6.2. (To change ft to m, multiply by 0.3048)

duration of the simulation, there is a substantial increase in the predicted excess temperatures. In fact, for the conditions represented in Fig. 5.3, an increase in simulation time would likely have resulted in predicted excess temperatures that violate state thermal standards. However, such a prediction is unimportant because the particular simulation then represents conditions so unrealistic that the results become irrelevant.

Although the problem of underprediction is inherent in all the applicant's results, it is less significant for the realistic cases. For conditions more realistic than those in Fig. 5.3, the predicted excess temperatures are sufficiently low so that no violations of thermal standards would be expected as a result of increases of simulation duration in the physical model. This expectation is confirmed by the staff's mathematical model study.

5.3.1.2 Staff's thermal analysis

The staff has performed an independent thermal analysis for the proposed operation of the once-through cooling system. Depth-averaged numerical models from the Unified Transport Approach[3] were used to simulate plant-induced flows, natural flow, and water temperatures. Predictions have been made for conditions typical of mid-July, since this is the time of year when thermal impacts should be the most severe. The modeled region is a rectangle measuring approximately 24,000 m (80,000 ft) in the longshore direction and approximately 12,000 m (40,000 ft) in the offshore direction. This region with the numerical grid system superimposed is shown in Fig. 5.5.

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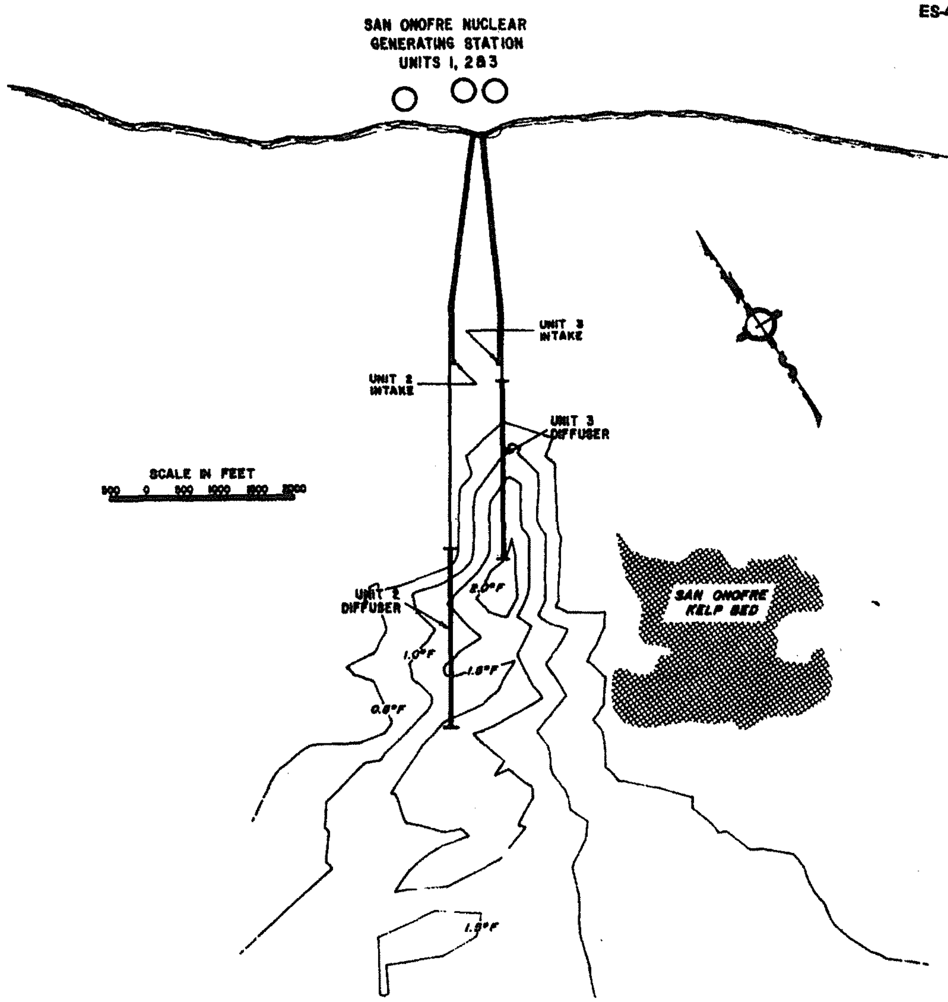

Fig. 5.3 Excess temperature at the surface predicted in the physical model study, for the case of no ambient flow. Source: ER Fig. 5.1-1. (To change ft to m, multiply by 0.3048; to change F° to C°, divide by 1.8.)

One numerical model was used to generate the induced flow from intakes and discharges from all three units. In this model, the intakes are represented as point sinks and the Unit 1 discharge is represented as a point source. The diffusers for Units 2 and 3 are each represented as a superposition of five jets. The hydrodynamics of each jet is modeled using a uniformly valid singular-perturbation theory, numerically corrected for bathymetry. The individual flows from the three intakes and discharges were summed to generate a total plant-induced flow field, as shown in Fig. 5.6.

A quasi-potential hydrodynamic model was used to generate the magnitude and direction of the natural currents and free surface displacement resulting from two tidal components and a net downcoast drift, at each grid element. The open-water boundary conditions were adjusted to produce flows which are consistent with observed data[4-7] from current meters and drogues. Three individual runs were executed, one for each of the two tidal harmonics (a 12.4 hr period and a 24.8 hr period), and a third to generate the drift current. These three flow components were combined, with the appropriate phase relationships, to produce a simulation of the natural flow field during mid-July conditions.

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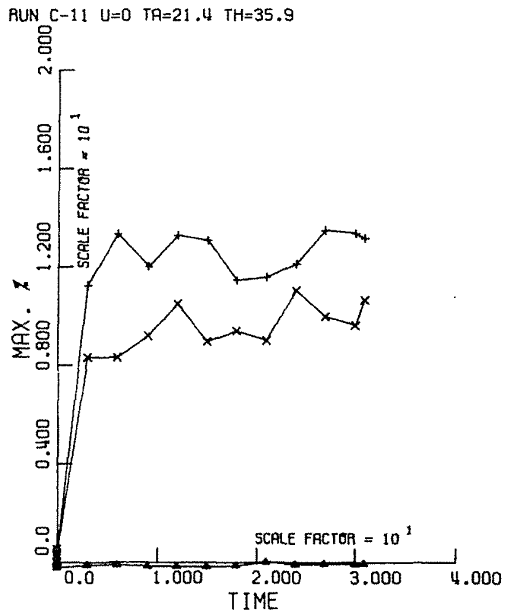

Fig. 5.4. Summary of maximum temperature excesses (in percent of source temperature excess) measured anywhere in basin (+), beyond 305 m (1000 ft) of diffusers (x), and ambient temperature (Δ) (Run C-11, u = 0.0 knot).

Water temperatures were computed using a depth-averaged thermal model. Inputs to this model were the calculated natural and plant-induced flows, along with meteorological parameters used for surface heat transfer calculations. The required meteorological variables are incoming solar radiation, cloud cover, air temperature, wind speed, and relative humidity. The incoming solar radiation is the mid-day value, which the code automatically adjusts for the time of day, from sunrise to sunset. The remaining parameters are taken to vary sinusoidally over one day and, therefore, require as input the daily average, the amplitude of the daily variation, and the time of maximum value. Typical values for these parameters during mid-July were used and are shown plotted as a function of time in Fig. 5.7.

This thermal model was first run without thermal output or flow from any of the units to produce a five-day simulation of ambient ocean temperatures. Subsequently, the calculation was repeated, with all three units operating at full capacity, to predict the total temperature field. These two results were then subtracted to generate excess temperature plots. Figures 5.8 through 5.15 show ambient flow and excess temperature plots at 6 hr intervals during the fifth day of the simulation at 2:00 am, 8:00 am, 2:00 pm, and 8:00 pm respectively. Isotherms are plotted in increments of 0.28°C (0.5°F) from 0.28°C (0.5°F) to 2.8°C (5.0°F). In general, the hottest spots occur directly above the discharge for each unit, with Unit 1 being consistently hotter than Unit 2 or 3. In addition, during the part of the tidal cycle when the natural flow is downcoast, there is a secondary warm spot approximately 3000 m (10,000 ft) downcoast of the discharges. This apparently is a result of the influence of the shape of the shoreline on the flow which, in turn, causes the plume from Units 2 and 3 to intersect the plume from Unit 1 at this point downcoast.

California thermal standards require that the 2.2°C (4°F) excess temperature isotherm never reach the shoreline or bottom, and that the 2.2°C (4°F) surface isotherm must be within 305 m (1000 ft) of the discharge point during at least one-half of the tidal cycle. Although the thermal model is depth averaged, it is still possible to address the state standards with the model results because the ambient crossflow has a destabilizing effect upon the discharge buoyancy. During portions of the tidal cycle, the ambient crossflow is of sufficient magnitude to dominate the stable stratification, resulting in mixing of the plume to the ocean bottom in the neighborhood of the diffuser. Recent work by Almquist[8] provides the basis for determination of conditions for vertical mixing. According to Almquist, the warm plume will mix to the bottom when the ratio of the ambient crossflow velocity to the cube root of the buoyancy flux per unit length of diffuser is greater than one. Figure 5.16 (a) is a plot of this stability parameter versus time for one tidal cycle based on the staff's ambient flow predictions. The shaded area shows the period during the tidal cycle when instability will occur and the water column will be vertically homogeneous.

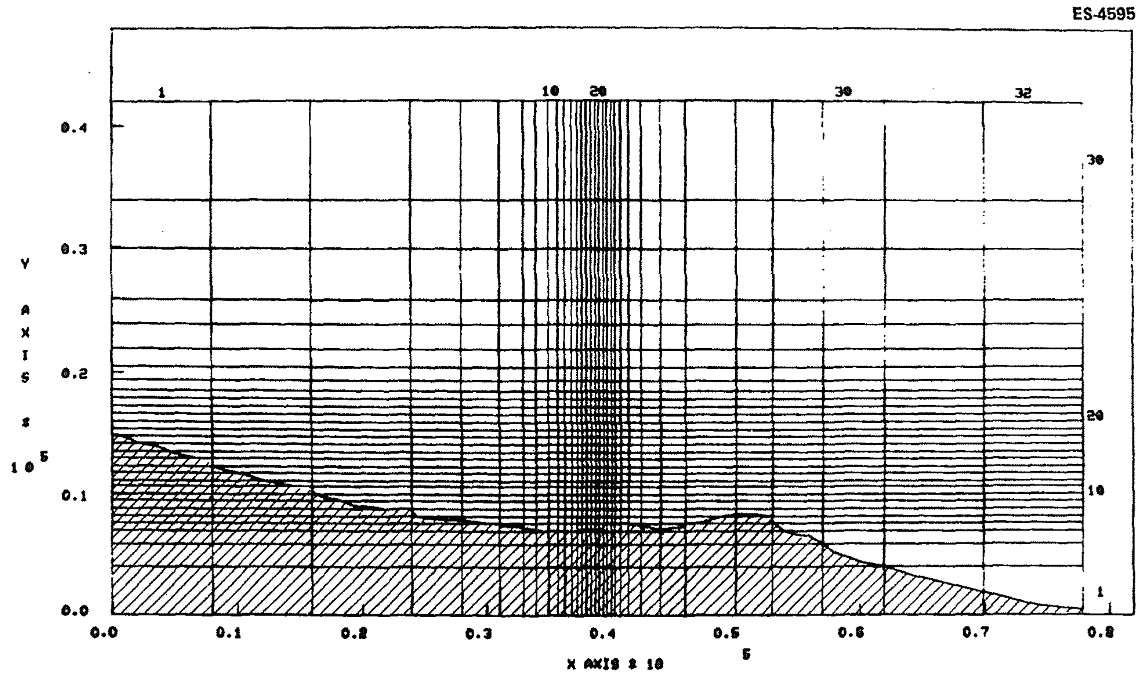

Fig. 5.5. Plot of region and grid system used for the mathematical model applications.

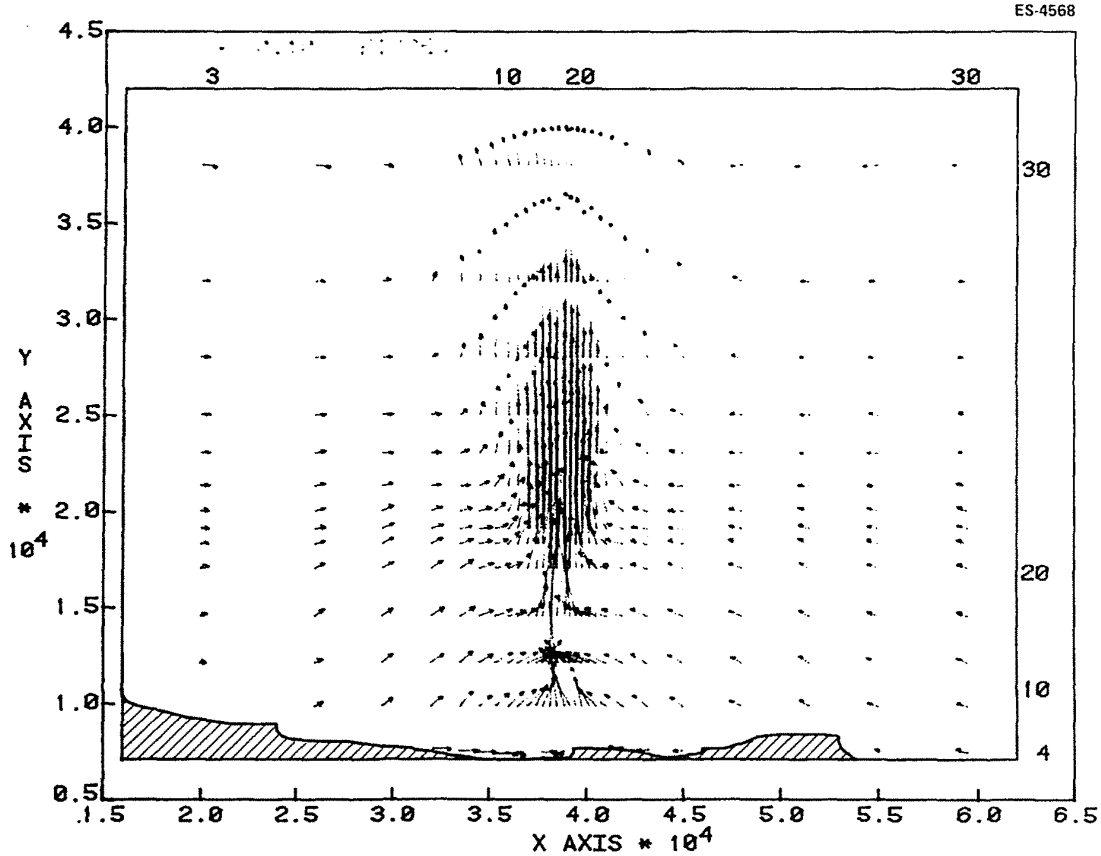

Fig. 5.6. Predicted, depth-averaged, plant-induced flow field for Units 1, 2, and 3.

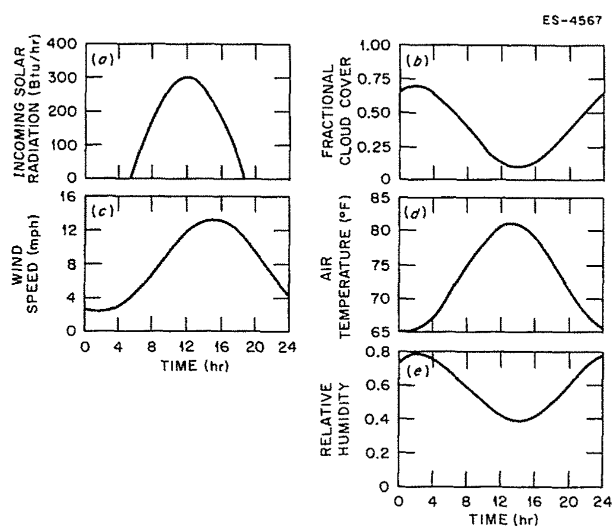

Fig. 5.7. Plots of meteorological variables as a function of time use in the thermal model. (To convert mi to km, multiply by 1.6; to convert °F to °C, subtract 32 and divide by 1.8.)

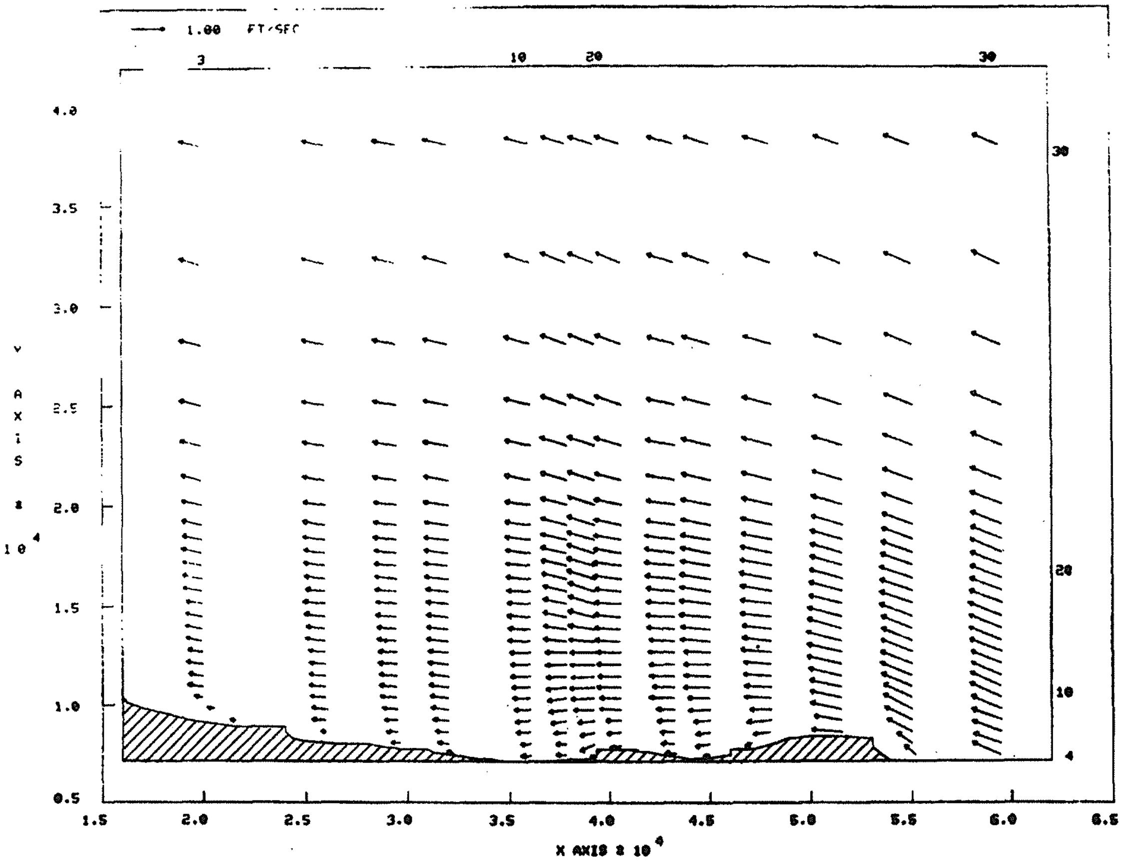

Fig. 5.8 Predicted natural flow field in the San Onofre region at 2:00 a.m. on the fifth day. (To change ft to m, multiply 0.3048; to change °F to °C, subtract 32 and divide by 1.8.)

Figure 5.16 (b) is a plot of the maximum excess temperature in the vicinity of the diffuser as a function of time for one tidal cycle. The shaded portion of this curve represents the period during the tidal cycle when the excess temperature is greater than or equal to 2.2°C (4.0°F) and the plume is vertically well mixed. In other words, the shaded area in this figure reflects the portion of the tidal cycle that will violate state thermal standards as applied to excess bottom temperature. It is clear from this figure that bottom excess temperatures greater than 2.2°C (4.0°F) are predicted to occur for two hours during the tidal cycle. Because, however, this prediction, based on a low ambient drift current, is conservative, excessive incremental bottom temperatures should not occur during each tidal cycle but rather during periods of worst case conditions.

With an assumed persistent drift, the data shown in Figs. 5.8 through 5.15 indicate that the constraints on the surface and shoreline excess temperature will be satisfied. The model is inadequate for addressing the issue of bottom temperature. However, at worst, the 2.2°C (4°F) excess temperature should only touch the bottom over a very limited area in the vicinity of the Unit 2 and 3 diffusers. On the basis of these results, the staff believes that violations of the state thermal standards are unlikely.

Heat treatment

Heat treatment will be necessary to control biological growth in the discharge conduits, intake conduits, and screenwells. Heat treatment consists of decreasing the flow rate through the heat-dissipation system while maintaining a constant waste-heat rejection rate. The result is an increased temperature rise across the condensers.

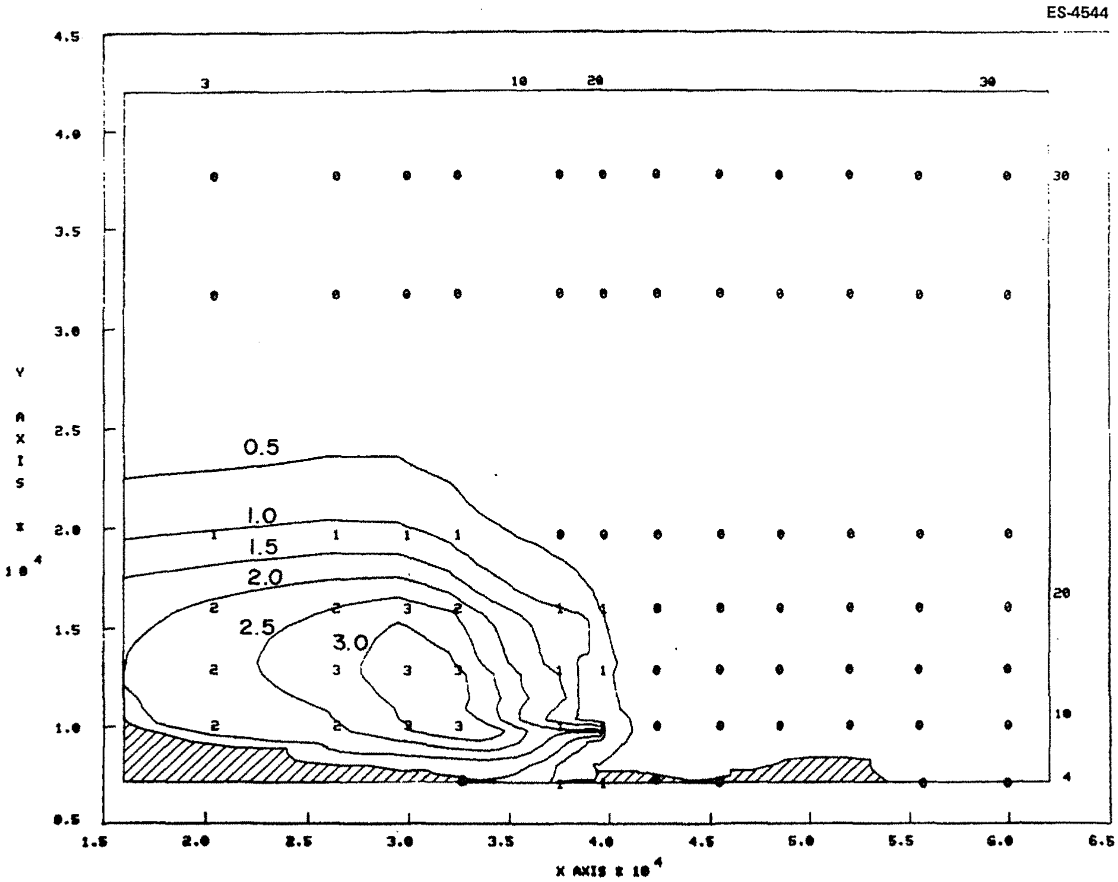

Fig. 5.9. Predicted excess temperatures in the San Onofre region at 2:00 a.m. on the fifth day. Isotherms are plotted in increments of 0.28°C (0.5°F) beginning with the 0.2°C (0.5°F) isotherm. (To change F° to C°, divide by 1.8.)

Discharge heat treatment will be required only when none of the following conditions are met:

1. discharge temperatures exceed 26.7°C (80°F) for a minimum of 1000 hrs,

2. discharge temperatures exceed 29.4°C (85°F) for 150 hrs, or

3. discharge temperatures exceed 32.2°C (90°F) for 31 hrs.

On the basis of these conditions it is expected that discharge heat treatment will be required only infrequently and usually during the winter. When discharge heat treatment is required, it will be performed at a discharge temperature of 40.6°C (105°F) for a duration of 1.1 hrs for Unit 2 and 0.9 hrs for Unit 3. During discharge heat treatment, discharge flow rates will be reduced and discharge temperatures will be increased. The discharge excess temperature will be the difference between the ambient water temperature and 40.6°C (105°F.) The reduction in the discharge flow rate will be proportional to the increase in the discharge excess temperature.

Although the exact nature of the thermal plume resulting from discharge heat treatment will be dependent upon the ambient conditions at the time of heat treatment, the thermal plume will be qualitatively similar to the plume resulting from normal operation as shown in Figs. 5.9, 5.11, 5.13, and 5.15. However, the flow is reduced and the temperature increased, so that the plume will be somewhat warmer and smaller in spatial extent than that from normal operation. The greatest plume temperatures will occur if Units 2 and 3 are heat treated simultaneously. A warmer plume will persist the longest when the heat treatment for these units are sequenced.

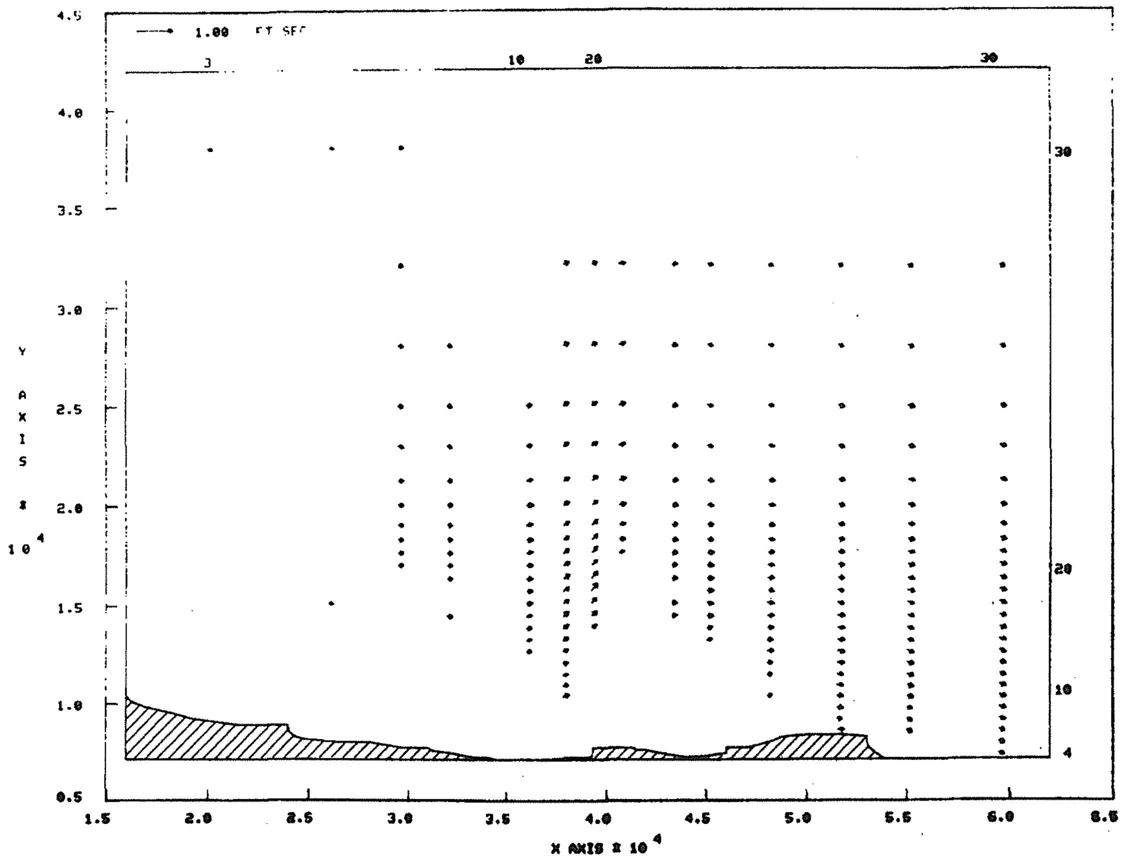

Fig. 5.10. Predicted natural flow field in the San Onofre region at 8:00 a.m. on the fifth day. (To change ft to m, multiply by 0.3048; to change F° to C°, divide by 1.8.)

During the summer months, discharge heat treatment should increase far-field plume temperatures by no more than 25% if both units are heat treated simultaneously (an unlikely event due to the increased probability of a reactor scram) and by no more than 15% if the units are heat treated sequentially. Plume temperatures at this extreme would persist for several hours, and plume temperatures would return to normal within several tidal cycles.

During the winter, the thermal plume should exhibit temperature distributions no greater that those predicted during the summer (Figs. 5.9, 5.11, 5.13, and 5.15). Excess temperatures during winter heat treatment will be greater than during the summer since a greater condenser temperature rise will be required to meet the design discharge temperature of 40.6°C (105°F). For an ambient water temperature of 10°C (50°F) (typical of winter) excess temperature at the San Onofre kelp bed would be approximately 4°C (7.2°F) if the Units 2 and 3 discharges are heat treated simultaneously and 2 to 3°C (3.6 to 4.8°F) if the discharges are heat treated sequentially.

Intake conduit and screenwell heat treatment will be performed by reducing the flow rate through the heat-dissipation system, thereby increasing the temperature rise across the condensers, and by reversing the flow direction so that ambient water is withdrawn through the diffuser and heated water is discharged from the velocity cap intake. The duration of this heat treatment will be 2.1 hr at an anticipated maximum temperature of 37.8°C (100°F). The plume produced by discharge through the velocity caps will resemble the thermal plume from Unit 1. Since this discharge does not induce the dilution produced by diffusers, the heat-treatment plume will be considerably hotter, though much smaller, than the plume resulting from normal plant operation. Plume temperatures will decrease approximately as the square of the distance from the intakes. Heat treatment on either the Unit 2 or the Unit 3 intake will have an indirect impact on the thermal plume of the unit operating normally. If, for example, the Unit 2 intake is heat treated while Unit 3 is operating normally, the Unit 2 heat treatment plume could be advected during certain times in the

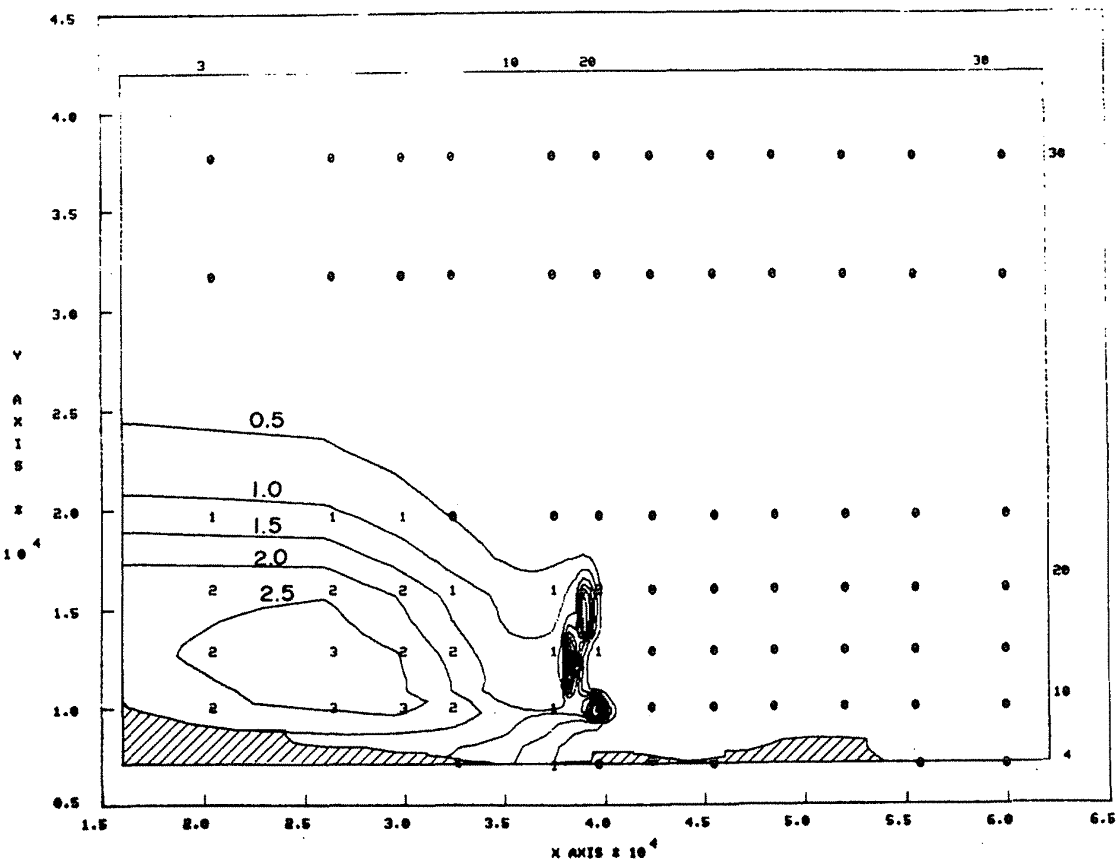

Fig. 5.11. Predicted excess temperatures in the San Onofre region at 8:00 a.m. on the fifth day. Iostherms are plotted in increments of 0.5°F beginning with the 0.5°F isotherm. (To change F° to C°, divide by 1.8.)

tidal cycle towards the Unit 3 intake. As a result, water at temperaures above the ambient could be drawn into the Unit 3 intake, resulting in a temperature rise in the Unit 3 discharge plume. Similarly, Unit 3 intake heat treatment could affect the plume from Unit 2. This recirculation phenomenon will be offset by virtue of the fact that only one unit will be discharging through the diffuser. Therefore, far-field diffuser plume temperatures will likely be less during intake heat treatment than during normal plant operations.

Both discharge and intake heat treatment will produce plumes showing temperatures greater than plume temperatures expected during normal operations. These increased temperatures will be greatest near the point of discharge, and will be of short duration returning to normal within several tidal cycles after completion of heat treatment.

Should it be determined that heat treatment results in significant excess temperatures at biologically sensitive areas, impacts could be mitigated by scheduling heat treatments during phases of the tidal cycle (such as periods when the tidal flow will transport the thermal plume away from areas of concern) that will minimize excess temperatures occurring in such areas.

5.3.2 Chemical discharges

The assessment of the effect of chemical discharges on water use contained in the FES-CP (5.2) is still, for the most part, valid. The discussion of the impacts of copper and nickel discharges has been altered by the change to titanium condenser tubes (3.2.4.1), and these discharges should not affect water use since the tubes no longer contain copper or nickel.

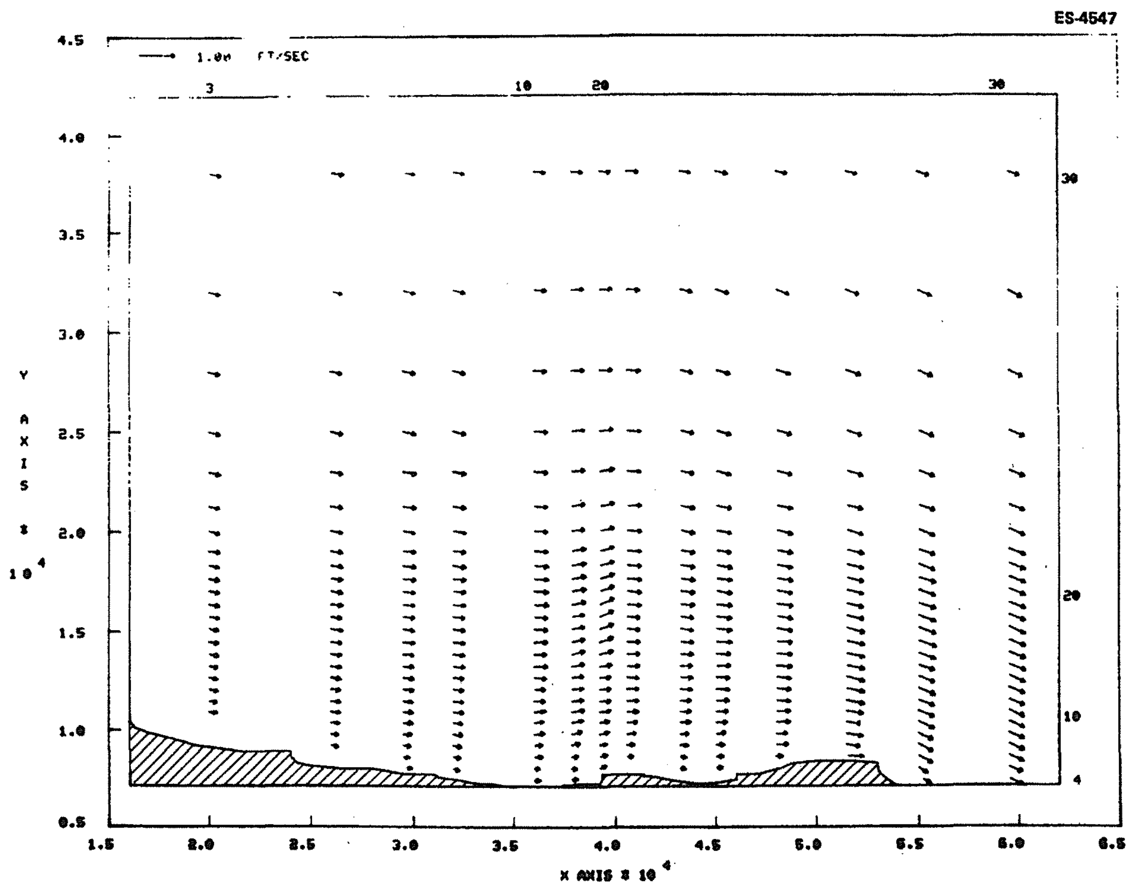

Fig. 5.12. Predicted natural flow field in the San Onofre region at 2:00 p.m. on the fifth day.

A National Pollutant Discharge Elimination System Permit for SONGS 2 & 3 was issued on June 4, 1976, by the California Regional Water Quality Control Board, San Diego Region. The chemical effluent limitations imposed by this permit are given in Sect. 3.2.4.1.

5.4 ENVIRONMENTAL IMPACTS

5.4.1 Terrestrial environment

Generally, operation of SONGS 2 and 3 and associated transmission lines should have no significant impact on the terrestrial ecological characteristics of the area. Although the transmission line routes have been modified since the issuance of the construction permit (3.2.5), the analysis of projected impacts as set forth in the FES-CP (5.3.1) remains the same. All new transmission lines will be constructed on existing rights-of-way; a total of 5.2 ha (12.8 acres) of land will be required for access road extensions and for new tower bases. The fire break which was bulldozed adjacent to the transmission line on Camp Pendleton Marine Base is expected to be maintained by periodic blading. Impacts associated with this operation should be minimal.

Other potential terrestrial impacts associated with operation of SONGS 2 and 3 which were not addressed in FES-CP are as follows. Some audible noise will be generated from the operation of the transmission lines. Noise levels, however, will be well within the urban evening levels accepted by the public (ER, Section 5.5.1). The transmission lines will be designed to minimize any affects on radio and television reception (ER, Section 5.5.1). Maintenance of the transmission lines (washing and repair work) requires that the access

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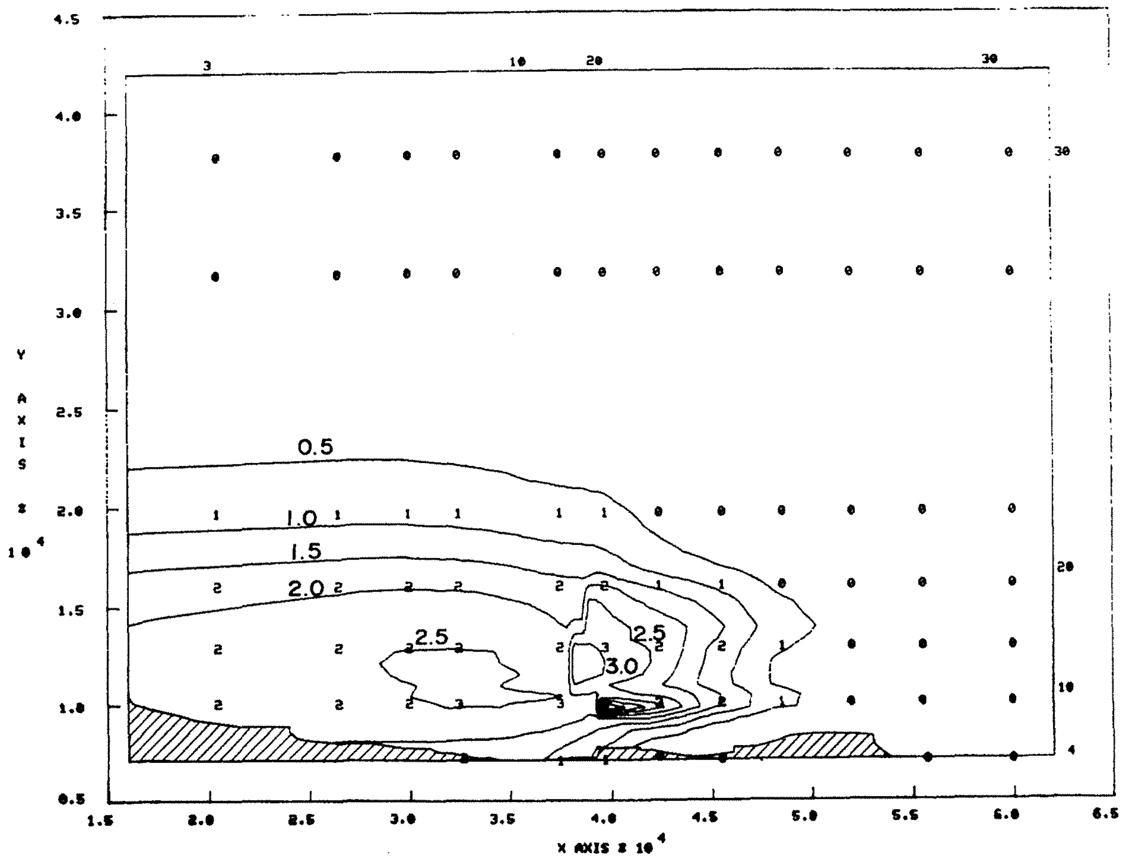

Fig. 5.13 Predicted excess temperatures in the San Onofre region at 2:00 p.m. on the fifth day. Isotherms are plotted in increments of 0.5°F beginning with the 0.5°F isotherm. (To change F° to C°, divide by 1.8.)

roads be kept in good condition by blading (ER, Suppl. 1, Item 21); associated impacts should be minimal. Maximum ground-level field gradients for all transmission lines will not exceed 7.5 kV/m (ER, Suppl. 1, Item 20). Generally, no harmful effects occur from the electrical fields associated with lines operating at 230 kV and below.[9]

5.4.2 Impacts on the aquatic environment

5.4.2.1 Effects of the heat dissipation system

A description of the heat dissipation system to be employed at SONGS 2 and 3 is found in Sect. 3.3 of the FES-CP. Design changes that have occurred since then are discussed in 3.2.2 of this statement. The only changes of potential significance for the assessment of biological effects involve the final specifications for the fish return system, the biocide use program, and the composition of the condenser tubing. Assessments of most major potential impacts also have been reevaluated in light of additional data obtained during technical specifications monitoring programs for SONGS 1 and from construction and preoperation monitoring programs for SONGS 2 and 3 (Section 2.5.2). Except as noted, the reassessments have resulted in the same conclusions that were reached in the FES-CP.

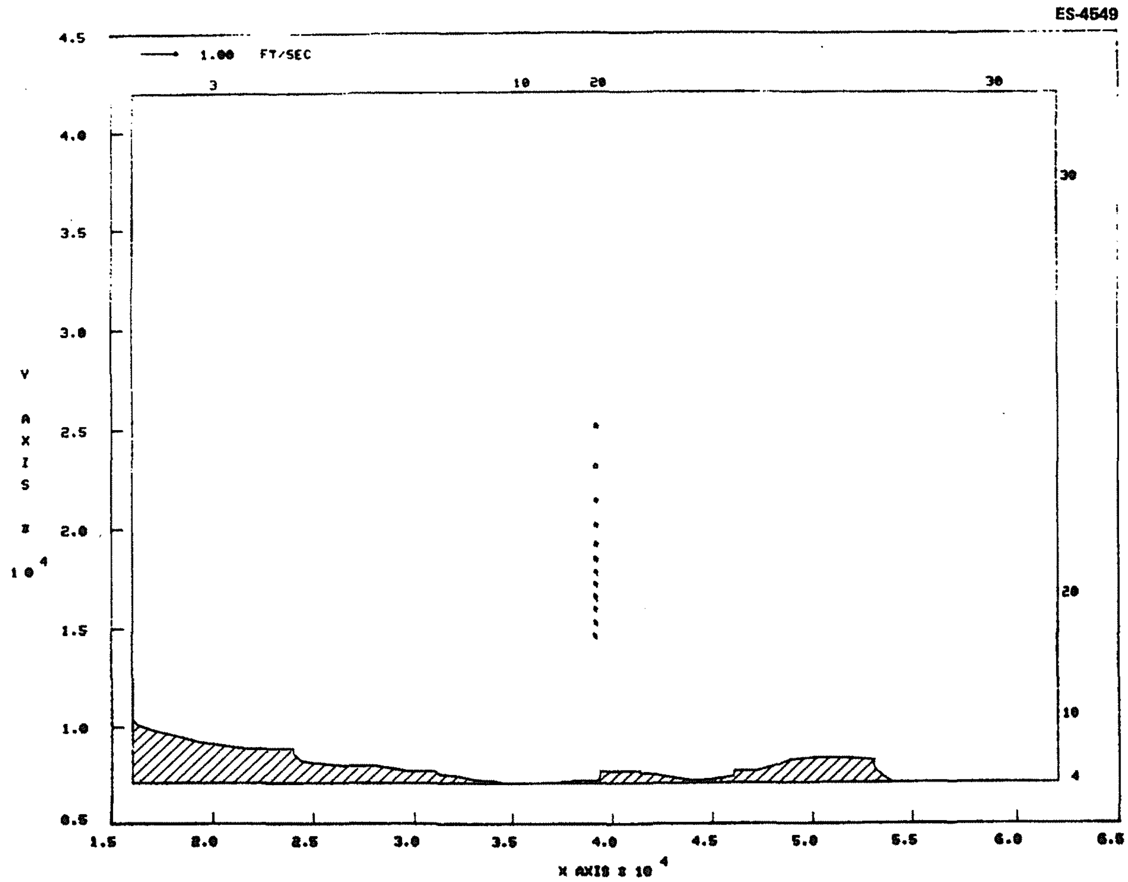

Fig. 5.14. Predicted natural flow field in the San Onofre region at 8:00 pm on the fifth day.

Thermal effects

The discharges from SONGS 2 & 3 must conform to regulations of the California State Water Resources Control Board, the Environmental Protection Agency (with regard to thermal discharges), and the California Regional Water Quality Board, San Diego Region, (under the auspices of the EPA) with regard to NPDES permit considerations (primarily chemical effluent limitations). The regulatory restrictions on thermal discharges are found in Sect. 5.1.1 of the ER; the NPDES permit, as amended, is found in Appendix 12C of the ER.

The results of thermal models used to evaluate temperature increases attributable to SONGS 2 & 3 (and incremental to SONGS 1) are discussed in Sect. 5.3.1. These data indicate that the thermal plume characteristics will be different from those estimated in the FES-CP and in the ER. Since the area to be affected by thermal discharges is now estimated to be greater than previously thought and since areas of substantial biological importance potentially will be affected (e.g., kelp beds), a reassessment is necessary.

Plankton. More planktonic organisms will be affected by thermal discharges than estimated in the FES-CP because the plume will cover greater area. The types of impact will, however, be the same (e.g., species compositon changes, greater respiration rates), and significant changes should be localized. The staff believes that changes which are produced in plankton communities will not threaten the ecological integrity of the near-shore region surrounding the facility (see pp. 5-26 to 5-32 of the FES-CP for a description of the anticipated effects).

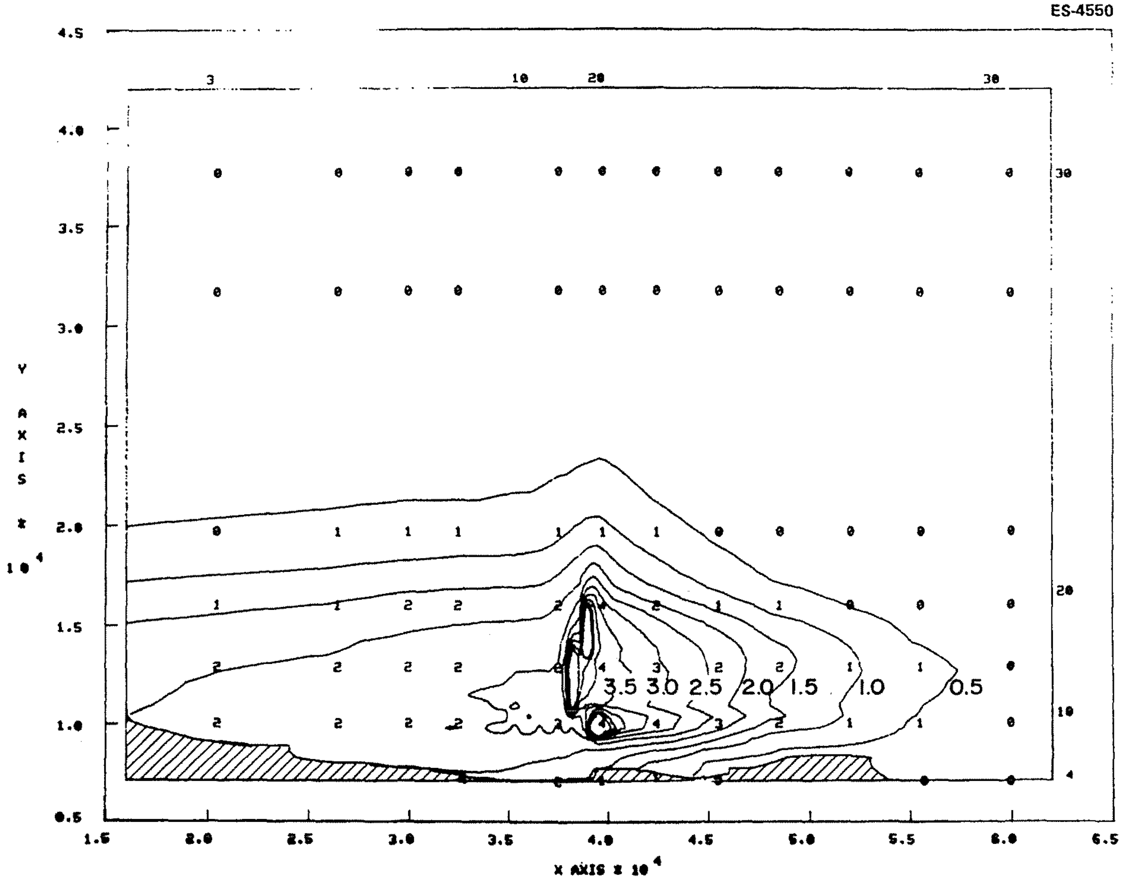

Fig. 5.15. Predicted excess temperatures in the San Onofre region at 8:00 p.m. on the fifth day. Isotherms are plotted in increments of 0.5°F beginning with the 0.5°F isotherm. (To change F° to C°, divide by 1.8)

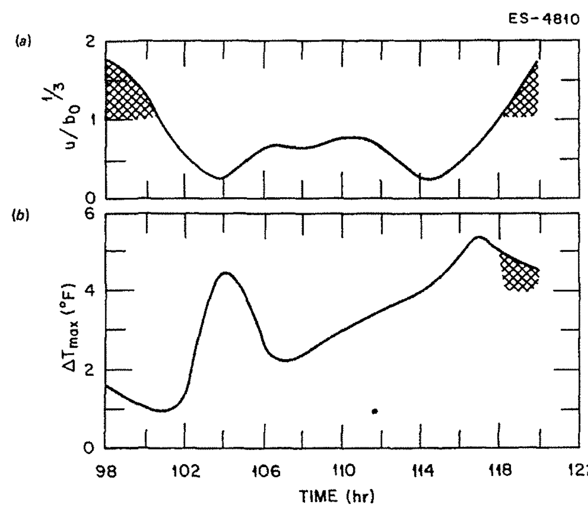

Fig. 5.16. (a) Plot of stability parameter versus time. The shaded area represents periods of vertical mixing. (b) Plot of maximum excess temperature versus time. The shaded area represents the period during which excess bottom temperatures are predicted to be greater than 4°F. (To change F° to C°, divide by 1.8.)

Fish. The types of impact on fish to be expected as the result of thermal discharges are the same as those discussed in the FES-CP. However, with more area to be influenced by the effluent, more fish potentially will be affected. The most observable change is likely to be shifts in the types of species (and their numbers) which inhabit the area; e.g., species which normally exhibit increased standing crops during naturally warm years will be more prevalent. Although the area of potential impact will be greater than estimated before, no fish populations are expected to be adversely impacted in the vicinity of the facility. Species composition changes, however, may affect commercial and recreational fishing within the thermal plume (in some cases adversely, and in others, beneficially; see FES-CP for details). However, because the plume will occupy a relatively small area of the available fishing space nearby, no significant changes in harvest rates for the various species are expected.

As stated in the FES-CP, cold kills of fish are not likely to occur to any large degree. The principal reasons are the relatively high ambient winter temperatures and the fact that all three units are not likely to be inoperative at any given time.

Benthic fauna. The major component of the ecosystem expected to receive the greatest impact from thermal discharges is the benthic community. Unlike free-swimming organisms, benthic individuals cannot easily avoid undesirable temperatures. And unlike planktonic organisms, they do not regenerate quickly to compensate for losses or experience continual, rapid recruitment from surrounding waters. Two major categories of the benthic community exist: animals, such as starfish and molluscs, and attached algae, the most conspicuous of which is kelp (discussed in the following section).

Among the benthic fauna recorded in the vicinity of SONGS during surveys conducted in 1977 in compliance with Environmental Technical Specifications criteria for SONGS Unit 1 were the gastropod molluscs Astraea undosa, Kelletia kelletii, and Roperia poulsoni, the asteroid echinoderm Pisaster giganteus, and the echinoid echinoderm Strongylocentrotus franciscanus.[10]

Although there have been only a limited number of detailed studies concerning the effects of temperature on marine species inhabiting the Pacific Coast, some recent laboratory simulation experiments of 12 to 14 weeks duration have examined the effects of thermal effluent on the survival, growth, and state of health of seven motile invertebrates from shallow rocky habitats along the southern California coast.[11] The treatment conditions simulated temperatures measured at distances of 84 and 335 m (276 and 1098 ft) from the cooling-water discharge structure of the Redondo Generating Station, located approximately 100 km (62 miles) upcoast of SONGS. Several of the species displayed low survival and impaired growth, especially among large adults, in response to the simulated thermal plume conditions at 84 m. Weekly mortality data for S. franciscanus, P. ochraceus, and R. poulsoni showed that individuals of all three species began to die when the temperature fluctuated over a range of 19° to 23°C (66° to 73°F), with a mean for the week of 21.4°C (70.5°F). No deaths had occurred the previous week when the same temperature range prevailed and the mean was slightly higher 22.8°C (73°F). The mortality observed during the second of these two weeks may, however, actually have been a delayed response to the higher average temperature of the previous week.

In the test involving R. poulsoni under a different thermal regime, deaths began occurring when the temperature flunctuated between 18° and 24°C (64° and 75°F) during the week, with a mean of 20.3°C (68.5°F). Although mortality began to appear at a lower mean temperature than in the previous experiment with this organism, the maximum temperature in this second experiment was 1°C (1.8°F) higher (24° vs 23°C) (75°F vs 73°F) and the temperature range was 2°C (3.6°F) wider (6° vs 4°C) (4.28° vs 39.2°F) than in the previous experiment. These results demonstrate the complicated nature of temperature effects; that is, adverse conditions can result from a critical high temperature of short duration, an extreme temperature fluctuation of short duration, or a prolonged period of a high but normally subcritical temperature.

The ambient depth-averaged temperatures predicted for the hottest time of the year (end of July) in the vicinity of SONGS are shown in 5.3.1. This section also contains data on the temperature expected during the operation of all three units. Temperatures potentially as high as 27.8°C (82°F) may occur naturally, and increases of 0.5° to 1.7°C (0.90° to 3.1°F) brought about by the operation of all three units can occur within an area of several square kilometers.

On the basis of the 1976 study,[11] the staff concludes that several components of the benthic fauna in the vicinity of SONGS would probably be adversely affected in areas where weekly mean temperatures of 22°C (71.6°F) prevail for one month or more or where daily temperatures reach or exceed 24°C (75°F). It is not, however, anticipated that temperatures averaging 22°C will occur for more than 2 to 3 weeks or that the area experiencing temperatures of 24°C or greater as a result of SONGS operation will be considerably larger than the area experiencing these temperatures under natural conditions.

The staff concludes that any impacts to the benthic fauna as a result of thermal discharges will be minimal and of an acceptable nature.

Kelp. Kelp beds off California occupy roughly 194 sq km (75 sq mi) of ocean bottom in water depths of 6-18 m (20-60 ft).[12] Although management efforts have possibly halted further severe decline, kelp bed coverage has decreased markedly since about 1920. Although this deterioration may have been partially a result of overharvesting, much of it is probably caused by the increased alteration of the near-shore environment by human activities. In particular, increased temperatures and increased turbidity have been shown to be inimical to kelp survival.[13]

Even without the influence of human perterbations, individual kelp beds experience long-term variations in stand density, productivity, areal extent, etc. Natural factors implicated in causing these variations include storm damage (causing detachment of plants), sand movement (burying holdfasts and causing detachment or prohibiting regeneration), introduction of turbid water masses, high natural temperatures, influx of grazing urchin masses, and fungal and bacterial diseases.[12] Thus, for example, in 1957-59, unusually warm temperatures off southern California caused an estimated loss of 90% of the regions' beds during this period (ER, pp. 2.2-28 and 2.2-29), as judged by surface examinations. Individual beds also commonly display changes in canopy extent during the year. For example, the three beds near the SONGS site showed marked variation in canopy area during 1975 and 1976 (Fig. 2.10). Typically, canopy tissue deteriorates during the warmest time of the year, leaving the basal portion of the plant (which is in cooler water) for regeneration when temperature and light conditions permit.[13] Reduced surface nutrients and higher bottom nutrient mixtures may also contribute to canopy deterioration and basal tissue regeneration respectively.[14]

Kelp beds represent a very important ecological community in California's near-shore waters. It has been estimated that kelp beds are at least three times more productive than the autotrophic components of other near-shore communities. Conservative estimates place the total standing crop of kelp in southern California at 1.8×10^9 kg (2 million tons) and new annual growth potential is on the order of 2-3 times this amount.[13] Kelp beds harbor numerous types of animals and plants, adding greatly to the diversity of an area. Invertebrates commonly found on the plants themselves include ostracods, copepods, amphipods, decapods, polychaetes, nematods, bryozoans, turbellaria and molluscs. Molluscs and echinoderms are kelp grazers prevalent on and around the plants. It is estimated that the larval, juvenile, and adult stages of 25 main sport fish use kelp beds for refuge and food gathering (eating the associated invertebrates, the kelp itself, or other algae), and the average standing crop of fish is estimated to be 300 kg/ha (300 lbs/acre).[13] Kelp not only enter the food chain via grazers, but they contribute large quantities of organic matter to the detritus-based food chains. For example, since several detritus feeders are intermediate in the grazing food chain of many of California's commercial fishes, kelp indirectly influences the populations of these fishes through the production of detritus.[13]

Kelp is an important commercial commodity as well. Although used extensively in the past for such diverse things as fertilizer, cattle feed, and for the production of potassium, acetone, and iodine, most kelp today is processed for the production of algin, a polysaccharide with numerous industrial uses.[12] It is estimated that roughly 15% of the annual kelp production is harvested yearly at a landed value (1964 dollars) of $2 million (market value is roughly 4 times this figure).[13] The kelp beds in the vicinity of SONGS are not now harvested.

Besides the necessity for a favorable physicochemical environment, kelp requires a solid substrate for attachment. Thus, the local distribution of kelp beds in an unperturbed area is largely substrate-dependent. Near the SONGS site, sandy bottoms are prevalent limiting the areas where beds can develop. Natural environmental fluctuations (e.g., higher-than-average temperatures) can virtually denude an area, but, since the casual phenomena are short-lived, kelp beds generally reestablish themselves quickly. However, anthropogenic disturbances frequently completely eliminate kelp beds in their sphere of influence because they generally are of long duration. Even chronic, low-level perturbations which only slightly decrease kelp production often cause the consumption by grazers to outpace new growth.[13]

The temperature tolerance of kelp is probably a reflection of a combination of factors, including physiological responses, susceptibility to disease, and susceptibility to grazing. It has been rather well established that temperatures above 18-20°C (64-68°F) cause deterioration of kelp, and the degree of degradation is directly related to the duration of the exposure to these temperatures. Increased surface temperatures caused by SONGS operation (all three units) would have the effect of extending the period of canopy absence. During the hottest time of the year, data in Section 5.3.1 suggest that the closest kelp bed (San Onofre bed) will experience an average surface temperature increase (over a 24-hr period) of 1.4°C (2.6°F); the range of temperature increase will be 0.6-2.2°C (1-4°F).

Although daily natural temperature variations of 1°C (2°F) are not uncommon in the area (ER, p. 2.2-28), they would not be continuous in nature and thus might not affect the bed

as severely as the continuous SONGS discharges would, where the thermal plume may impinge on the bed for a longer time. Prediction of the degree to which canopy disappearance would be prolonged is impossible. Regeneration would be quicker in years with naturally cooler ocean temperatures, assuming the regenerative tissues remained unaffected (see below).

The greatest threat of SONGS to the long-term survival of the San Onofre kelp bed is the possibility of injury to the basal tissues from which the canopy is regenerated each year as the waters cool. Estimates for bottom temperatures within the bed at the end of July (Section 5.3.1) indicate that temperatures could reach 23-25°C (74-76°F), with a 24-hr mean of 24°C (75°F). Such temperatures would represent a 1-1.5°C (2-3°F) increase above ambient conditions encountered during the hottest portion of the year (conditions which are likely to persist for up to approximately a one-week period) (Section 5.3.1). Although the ambient temperatures given above would in and of themselves be detrimental to the kelp, exposure to them for up to a week would not likely cause permanent degradation of the entire bed[13] because the mean exposure temperature does not quite exceed a recognized threshold temperature for rapid degradation (24°C) and deeper portions of the bed would be slightly cooler than the average and would have a greater probability of maintaining a viable population. However, adding 1-1.5°C to these ambient temperatures could place the bottom kelp tissues in a critical temperature environment subjecting the tissues of most of the plants to temperatures greater than their short-term tolerance, and prolonging the period of time in which the plants would experience temperatures greater than 20°C (68°F), which would cause them to be more susceptible to grazing pressure, diseases, etc., leading to their eventual demise.[13] Since ambient bottom temperature in the region from August – early September may typically range up to 19°C (66°F) (Section 5.3.1), a several week period could exist in which temperatures exceed 19°C.

The information above suggests that the thermal discharges from SONGS 1, 2 and 3 may result in the destruction of at least a portion of the San Onofre Kelp Bed during the summer months. Under average conditions, the result may not be detectable or it may be manifested in a noticeably earlier decline of the canopy. However, under extreme worst case conditions (e.g., several days with high ambient temperatures and slack currents, and with all three plants operating continuously), destruction of the basal regenerative tissues might result. Although recolonization of the area from outside sources could occur during the cooler months, the community, if destroyed frequently, could never achieve a stable state characteristic of other kelp beds in the area. Furthermore, constant temperature increases coupled with added turbidity would be inimical to interim reestablishment since these factors tend to increase the effects of grazing.[13] The perennial occurrence of worst case conditions seems highly unlikely (Section 5.3.1) and the staff thus concludes that the long-term thermal impacts from normal station operation are not likely to be severe. However, in view of (1) the potential additive of synergistic effects of turbidity and sediment with thermal discharges, (2) the ecological importance of kelp beds and their already diminished stature, and (3) the fact that the San Onofre bed represents about one-third of this resource along approximately 16 km (10 mi) of shoreline in the vicinity of SONGS, the staff recommends monitoring to ensure the bed's protection.

Heat treatment

In addition to the thermal discharge associated with the normal operation of the facility (see above), the applicant proposes to heat treat portions of the intake and discharge systems to remove biological growth (see Section 5.3.1.2). This antifouling procedure will result in periodic discharge temperatures higher than those normally encountered. As a result, the state required the applicant to perform a demonstration to determine if significant impacts will result from the procedure. This demonstration, in part provided for under part 316(a) of the Federal Water Pollution Control Act of 1972, was used to determine if the proposed process is acceptable to these government agencies. To date, approvals have been obtained from the California State Water Resources Control Board (Resolution No. 80-95 adopted December 18, 1980), thus removing any regulatory obstacles from the state for conducting the antifouling process.

As stated in Section 5.3.1.2, biofouling control will be needed primarily in the winter; ambient summer temperatures will normally be sufficiently high to obviate the need for the procedure at that time. Additionally, the state has imposed a five-week minimum treatment interval for each unit. Hence, the biological effects will be a manifestation of short-term intermittent stress. Localized mortality and chronic debilitation are inevitable, particularly for sessile organisms. However, only one community of organisms is judged to be significantly vulnerable ecologically – the San Onofre Kelp Bed.

The thermal effects of normal operation on kelp are discussed above along with more detailed information on thermal tolerances, etc. Since intake heat treatment should produce smaller far-field ΔT's than that produced by normal operation (Section 5.3.1.2), the effects on kelp will be less than or equal to the effects induced normally. Discharge heat treatment is

judged to produce potentially greater far-field thermal effects, however. Without dispersing currents (i.e., during a slack in the tidal cycle), kelp bed temperatures during the summer may increase by ca. 0.4°C (0.72°F) (above normal operations) (Section 5.3.1.2). This negligible increase would not be likely to affect the kelp, particularly since the canopy will be naturally reduced (see kelp discussion above) and the heated water is not likely to be near the bottom.

Discharge heat treatment during the winter may cause a temperature increase in the kelp bed of up to 4°C (7.2°F). The kelp are ordinarily tolerant of the absolute temperatures this would produce, but the rapid heat-up involved (e.g., 0.5 h) could be deleterious since the kelp would not be "hardened" for such a temperature regime. However, it is not possible to tell from the literature the severity of such an event. The plants could be only temporarily taxed physiologically and rebound without sequelae. Conversely, the stress could initiate an increased vulnerability to other, natural stresses such as predation, sloughing, and encrustation. Overt mortality is unlikely. In the absence of definitive data, it would be wise to (1) ensure continuation of the kelp monitoring program and (2) attempt to avoid heat treatment during unfavorable ocean current conditions. As pointed out in Section 5.3.1.2, effects can be mitigated by staggering heat treatment at Units 2 and 3 (thus allowing thermal dispersion from the first treated unit before treating the second) and by conducting the antifouling procedures when current and tidal cycles are known to move the adjacent water mass away from the kelp bed.

Turbidity and sediment transport effects

The FES-CP discusses the types of effects turbidity increases due to SONGS operation will have on the various biological communities, indicating that it is not possible to predict the areal extent of this impact.

The organisms likely to receive the greatest impact from increased turbidity are those which cannot readily avoid adverse conditions or do not regenerate quickly (or experience rapid recruitment from surrounding waters), namely, the benthos. Since the San Onofre Kelp Bed is estimated to be enveloped within the thermal plume, it is likely that it will also experience increased turbidity. The effect on the kelp would potentially be decreased photosynthesis, possibly causing many of the plants to die if the exposure is continuous (a 1% increase in the absorption coefficient has been found to result in a 20% loss in net photysynthesis at 15 m (49.2 ft)[13] and burial of the holdfasts in particles which settle out, inhibiting regeneration and recolonization. Regardless of the magnitude of these effects, their presence would add to the probability that the kelp bed would be adversely affected (see preceding section).

Some of the effects of increased sediment transport on benthic fauna are addressed in the FES-CP. The staff has further addressed the impact of the change in sediment size in areas near the SONGS site which would result from sediment redistribution. A study conducted during SONGS 1 operation, shutdown, and subsequent startup showed a significant reduction in the number of species and the total abundance of individual benthic fauna (primarily molluses and polychaete worms) within 200 m (656 ft) of the intake and discharge structure, probably because of the coarsening of the grain size of the sediments in this area.[15] Sediment coarsening appears to be mainly a result of the discharge of shells and shell fragments of fouling organisms (barnacles, molluscs) sloughed from the insides of the intake and discharge pipes during normal operation and especially during heat treatment.

The sediment-altered area associated with SONGS 1 (following 13 years of operation) is estimated to be approximately 125,600 m^2 (.048 mi^2), on the assumption of a circular pattern of effect with a radius of 200 m (656 ft).[15] Assuming sediment alteration associated with SONGS 2 and 3 forms a rectangular pattern approximately 200 m from the sides and ends of each diffuser, the area affected by SONGS 2 and 3 would be approximately 0.8 km^2 (0.31 mi^2). Adding this to the area affected by SONGS 1 (125,600 m^2 (0.48 mi^2)) plus an estimate of the area affected by heat-treatment backflushing of the SONGS 2 condenser (59,900 m^2 (0.023mi^2)) gives a total area affected by all three units, from both normal operation and heat treatments, of approximately 1.0 km^2 (.386 mi^2).

It is difficult, however, to extrapolate from the effects associated with the point source discharge of SONGS 1 to the 762-m (2500-ft) long dual, staggered diffusers of SONGS 2 and 3. SONGS 2 and 3 jointly are expected to have 5 times the cooling water flow rate, 3.3 times the intake pipe area per intake structure, and 12.5 times the total fouling surface area associated with the two outfall lines that SONGS 1 has.[15] None of these factors has been taken into consideration in calculating the area potentially affected by SONGS 2 and 3. The magnitude of the effect will also increase with duration of operation.

In contrast to the above prediction of benthic impoverishment, the staff concludes that a zone of enhanced species diversity and abundance is to be anticipated beyond the area of

sediment modification. This conclusion is also based on results of the Marine Review Committee study,[15] which indicates that within a zone of 200 to 800 m (656 to 2424 ft) from the intake and outfall of SONGS 1, diversity and abundance of benthic fauna show a positive correlation with proximity to these structures. It has been estimated that this area contains 2 times the diversity and 8 times the abundance of benthic fauna as the sediment-altered area within the 200-m (656-ft) radius of the outfall. This phenomenon is believed to be a result of organic enrichment from sinking plankton fragments and/or material continually resuspended by the localized turbulence of the discharged cooling water.[15]

Assuming an eliptical ring pattern for this area of enhancement, starting from a point 200 m (656 ft) on either side of the intake and outfall structures of SONGS 1, to 1200 m (3936 ft) upshore and downshore (the extent of enhancement appears to diminish between 800-1500 m (2624-4920 ft) downcoast) and extending for a distance of 400 m (1312 ft) beyond the 200-m (656-ft) point in the onshore and offshore directions (offshore/onshore effect is much less than longshore), the area of enhancement is estimated to be approximately 2.1 km^2 (0.81 mi^2).

Predicting the magnitude of an enhancement effect associated with SONGS 2 and 3 on the basis of SONGS 1 observations is complicated. The total volume of dead plankton dispersed might be approximately 5 times that of SONGS 1 as a result of the 5-fold increase in cooling water flow rate. However, the volume of discharge for each diffuser port is less than for the single outfall of SONGS 1 so that the distance the entrained plankton are dispersed would be expected to be less. There may also be considerable differences between the shallow current patterns where the SONGS 1 outfall is located and the current patterns in the deeper waters where the SONGS 2 and 3 diffusers will be located.

If it is assumed that the dispersal distances for dead plankton will extend approximately half the distance from the sediment-altered area surrounding the SONGS 2 and 3 diffusers as was found associated with the SONGS 1 discharge, and accounting for overlap, the area of enhancement would be approximately 2.4 km^2 (0.93 mi^2). Adding to this the area affected similarly by SONGS 1 gives a total of 4.5 km^2 (1.74 mi^2). This is an area approximately 5 times that estimated to show a reduction in benthic diversity and abundance. The staff concludes that the impacts likely to occur to the benthic fauna as a result of sediment transport effects are acceptable.

Entrainment

The staff's analysis of entrainment effects in the FES-CP remains valid (FES-CP, p. 5-7 to 5-12). A program on the mortality experienced by entrained ichthyoplankton is being planned currently at SONGS 1 and is expected to be submitted to the NRC staff in 1981. The results of this program should help to determine the significance of any impacts although the analysis presented in the FES-CP indicates that impacts should not be significant. The completion date for this study will be approximately one year after it is initiated.

The circulation of water from near-shore areas to offshore areas will cause some redistribution of species, particularly zooplankton, since species composition is not exactly the same for both areas (Section 2.5.2). Although this may result in long-term species composition changes, the areas affected should be small (FES-CP, Section 5.3.2) relative to the coastal areas as a whole around San Onofre. Because no other power plants or industrial facilities that could exert a similar influence exist within several miles, this impact is judged acceptable.

Impingement

The basic impingement analysis contained in the FES-CP remains valid. Some additional information is available, however, on the design and efficiency of the fish return system. The system is described in detail in Section 3.4 of the ER and in Section 3.2.2 of this document. Basically, the fish return system consists of a mechanism for shunting any fish entrained in the intake to a side holding area by means of an angled conduit design to avoid impinging them on the trash removal mechanisms in front of the final intake. Preliminary experimental results (ER, p. 5.1-20) indicate that perhaps 90% or more of the fish can be returned to the ocean unharmed. However, precise figures on the effectiveness of this system will not be available until the fish return system is in full-scale operation. The FES-CP analysis assumes a worst-case situation in which the fish return system is not at all effective. Under these conditions, 33 to 91 tonnes (36 to 100 tons) of fish per year would be removed from the San Onofre area. These figures are based on extrapolations from data obtained on SONGS 1 operation; new data do not indicate that these figures should be adjusted significantly. The majority of the fish impinged at SONGS 1 are queenfish, and, for reasons given in the FES-CP, losses from all three units should not have a significant impact on the population. Moreover, of the dominant recreational fish impinged at SONGS 1, losses were less than 0.8% of the amount taken by fishermen. Likewise, the primary

commercial fish of the area – jack mackerel, Pacific bonito, and white seabass – were seldom entrained at SONGS 1.

Offshore current induction

The analysis of the effects of induced circulation as given in the FES-CP (p. 5-16) remains valid, despite the design changes described in Section 3.2.2.

5.4.2.2 Effects of biocides and other chemical discharges

The FES-CP expressed concern about the potential long-term effects of copper being released into surrounding water by corrosion of the condenser tubing. Design changes have eliminated the plan to use a copper-nickel alloy for condenser tubing; titanium tubing will be used. Therefore, copper- or nickel-induced stresses to the receiving water from condenser tubing would not occur.

The FES-CP conclusion that the effects of chlorine will not be significant remains valid. However, new information is available on this subject. The applicant estimates that the effluent chlorine concentrations will be no greater than 1.5 ppm as total residual before discharge to the ocean (ER, p. 5.3-2). With a 10-to-1 mixing in the immediate vicinity of the diffuser ports (ER, p. 5.3-2), this value would be reduced to 0.15 ppm. The FES-CP required, and the applicant agreed, that the total residual concentration of chlorine and other halogens in the immediate vicinity of the discharge from each unit be limited to less than 0.1 ppm for no more than six 15-min periods each day [FES-CP, p. iv, item 7.a(2)]. Experience at SONGS 1 indicates that total residual chlorine concentrations quickly dissipate to undetectable quantities within a hundred or so meters of the outfall and, for any given 15-min dosing period, are only detectable over the outfall for 2 to 18 min (ER, p. 5.3-2). Even assuming a worst-case condition for SONGS 2 and 3 in which chlorine remains at levels around 0.15 ppm (total residual) in the vicinity of the outfall ports for as long as 30 min, any significant impacts are unlikely.[16] Thus, any chlorine effects are likely to be minimal and of an acceptable nature. Moreover, the difference in effect between discharges of 0.1 and 0.15 ppm are negligible. In view of this and in light of the provisions of the Federal Water Pollution Control Act Amendments of 1972, the staff does not believe that a more stringent limitation on chlorine discharges is necessary.

Miscellaneous chemicals will be discharged through the circulating water outfall system and will include laboratory wastes, ion exchange regeneration chemicals, and pH adjusters (Section 3.2.4 of this document and Section 3.5 of the FES-CP). The FES-CP analysis of the impact of these chemicals remains valid; that is, because of the small quantities involved, the great dilution factors present, and the relatively innocuous nature of most of these chemicals, impacts will not be detectable.

5.4.2.3 Effects of sanitary waste discharge

The effects of sanitary waste discharge are not discussed specifically in the FES-CP. However, any effects will be insignificant for the following reasons.

1. On the average, only about 26 m^3/day (7000 gpd) of secondary treated sewage will be discharged.

2. The discharge will be made into the circulatory water system at the rate of 0.02 m^3/min (5 gpm). The cooling water flow is about 1200 m^3/min (320,000 gpm). Thus, a 6400 dilution factor will result.

3. The resulting concentrations of suspended solids, BOD, N, P, coliform bacteria, and chlorine will not result in detectable incremental increases above ambient levels even before discharge into the ocean.

5.5 RADIOLOGICAL IMPACTS

5.5.1 Radiological impact on man

The impact on man associated with the routine release of radioactive effluents from SONGS 2 and 3 has been estimated. The quantities of radioactive material that may be released annually from the plant are estimated based on the description of the radwaste systems given in the applicant's ER and PSAR and using the calculational model and parameters described in NUREG-0017.[17] Using these quantities and site environs information, the dose commitments to individuals are estimated using models and considerations discussed in detail in Regulatory Guide 1.109. Additional assumptions and models described in Appendix B of this environmental statement were used to estimate integrated population doses.

5.5.1.1 Exposure pathways

The environmental pathways that were considered in calculating the radiological impact are shown in Fig. 5.17. Calculations of radiation doses to man at and beyond the site boundary were based on the radioactive material quantities shown in Tables 3.2 and 3.3, on site meteorological and hydrological considerations, and on exposure pathways at SONGS 2 & 3.

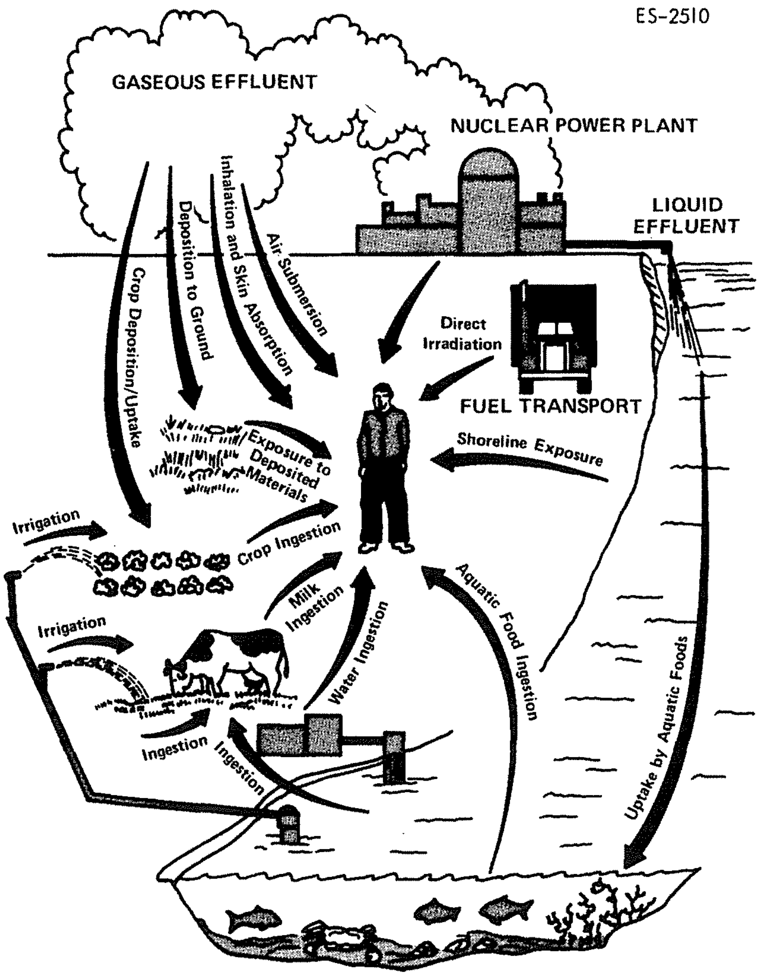

Fig. 5.17. Exposure pathways to man.

In the analysis of all effluent radionuclides released from the plant, tritium, carbon-14, radiocesium and radiocobalt inhaled with air and ingested with food and water were found to account for essentially all total-body dose commitments to individuals and the population within 80 km (50 miles) of the plant.

5.5.1.2 Dose commitments from radioactive releases to the atmosphere

Radioactive effluents released to the atmosphere from SONGS 2 & 3 will result in small radiation doses to the public. NRC staff estimates of the expected gaseous and particulate releases listed in Table 3.3 and the site meteorological considerations discussed in Sect. 2.4 of this statement and summarized in Table 5.1 were used to estimate radiation doses to individuals and populations.

Table 5.1. Summary of atmospheric dispersion factors and deposition values for selected locations near SONGS 2 & 3[a]

Location	Source[b]	X/Q (sec/m^3)	Relative deposition (m^{-2})
Nearest site land boundary (0.36 mile NNW)[c]			
	A	5.4 E–5	2.1 E–7
	B	2.4 E–5	9.3 E–8
Nearest residence and garden (1.3 mile NNW)[c]			
	A	4.8 E–6	2.0 E–8
	B	1.7 E–6	6.9 E–9

[a]The doses presented in the following tables are corrected for radioactive decay and cloud depletion from deposition, where appropriate, in accordance with Regulatory Guide 1.111, Rev. 1, "Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light Water Reactors," July 1977.

[b]Source A is gas decay tank, 48 purges per year, 12 hr per purge; source B is continuous release.

[c]"Nearest" refers to that type of location where the highest radiation dose is expected to occur from all appropriate pathways.

[d]Here E-x is used to indicate the factor 10^{-x}; i.e., 5.4 E–5 = 5.4 X 10^{-5}

(To change mi to km, multiply by 1.609.)

Dose commitments to individuals and the population can be estimated using different methodologies. The staff's assessment of dose is based on a 50-year commitment and is described in Regulatory Guide 1.109. The results of the calculations are discussed below.

Radiation dose commitments to individuals

The predicted dose commitments to the "maximum" individual from radioiodine and particulate releases are listed in Tables 5.2 and 5.3. The maximum individual has been estimated to receive the highest dose commitment from SONGS 2 & 3 and is assumed to consume well above average quantities of the foods considered (see Table A-2 in Regulatory Guide 1.109). The maximum annual air, total body, and skin doses from noble gas releases are presented in Tables 5.3 and 5.4.

Table 5.2. Maximum annual dose commitments to an individual near the SONGS 2 & 3 plant caused by particulate and liquid effluents

Location	Pathway	Dose (millirems per year per unit)		
		Total body	Thyroid	Other organs (if greater than 10% of dose)
	Iodine and particulate doses			
Nearest residence and garden (1.3 NNW)[a]	Ground deposit	0.66	0.66	NA
	Inhalation	0.07	0.48	
	Vegetation	0.40	2.5	
Totals		1.1	3.7	
	Liquid effluent doses			
Nearest fish	Fish ingestion	0.019	0.018	0.0016
	Invertebrate ingestion	0.0058	0.025	0.104
	Shoreline use	0.039	0.039	0.039
Totals		0.064	0.082	0.15

[a]"Nearest" refers to the location where the highest radiation dose to an individual from all applicable pathways has been estimated.

(To change mi to km, multiply by 1.609.)

Table 5.3. Maximum calculated dose commitments to an individual and the population from SONGS 2 & 3[a]

	Appendix I Design objectives	Calculated doses
	(Annual dose per reactor unit)	
Maximum individual doses		
Liquid effluents		
Dose to total body from all pathways, millirems	3	0.064
Dose to any organ from all pathways, millirems	10	0.15
Noble gas effluents (at site boundary)		
Gamma dose in air, millirads	10	4.6
Beta dose in air, millirads	20	14
Dose to total body of an individual, millirems	5	2.8
Dose to skin of an individual, millirems	15	8.5
Radioiodines and particulates[b]		
Dose to any organ from all pathways, millirems	15	3.7

Population doses within 80 km (50 miles)

	Total body (man-rems)	Thyroid (man-rems)
Natural radiation background[c]	700,000	
Liquid effluents	0.17	0.14
Gaseous effluents	21	46

[a]Appendix I design objectives from Sects. II.A, II.B, II.C, and II.D of Appendix I, 10 CFR 50; considers maximum doses to individuals and population per reactor unit. Source: *Federal Regist.* 40, 19442, May 5, 1975.
[b]Carbon-14 and tritium have been added to this category.
[c]"Natural Radiation Exposure in the United States," U.S. Environmental Protection Agency, ORP-SID-72-1 (June 1972); using the average State of California background dose of 97 millirems per year and year 2000 projected population of 262 million.

Table 5.4. Annual total-body, skin, and air doses at the nearest site boundary of SONGS 2 & 3 caused by gaseous radioactive effluents[a]

Location	Dose (millirem per year per unit)			
	Total body	Skin	Gamma air dose	Beta air dose
Nearest site boundary (0.36 mile WNW)[a]	2.5	8.3	4.2	14

[a]"Nearest" refers to that site boundary location where the highest radiation doses caused by gaseous effluents have been estimated to occur.
(To convert mi to km, multiply by 1.6.)

Radiation dose commitments to populations

The calculated annual radiation dose commitments to the population within 80 km (50 mi) of SONGS 2 and 3 from gaseous and particulate releases are presented in Table 5.3. Estimated dose commitments to the U.S. population are presented in Table 5.5. Background radiation doses are provided for comparison.

Within 80 km of the plant site, specific meteorological, populational, and agricultural data for each of 16 compass sectors around the plant were used to evaluate the doses. Beyond 80 km, meteorological models were extrapolated by assuming uniform dispersion of noble gases and continued deposition of radioiodines and particulates until no suspended radionuclides remained. Doses were evaluated using average population densities and food production values discussed in Appendix B. The doses from atmospheric releases during normal operation represent an extremely small increase in the normal population dose from background radiation sources.

Table 5.5. Annual total-body population dose commitments in the year 2000

Category	U.S. population dose commitment for the site
Natural background radiation, man-rems per year[a]	27,000,000
SONGS 2 & 3 operation, man-rems per year per site	
Plant workers	2600
General public	
Gas and particulates	160
Liquid effluents	<1
Transportation of fuel and waste	14

[a]Using the average U.S. background dose of 102 man-rems per year and year 2000 projected U.S. population from "Population Estimates and Projections," Series II, U.S. Department of Commerce, Bureau of the Census, Series P-25, No. 541 (February 1975).

5.5.1.3 Dose commitments from radioactive liquid releases to the hydrosphere

Radioactive effluents released to the hydrosphere from SONGS 2 & 3 during normal operation will result in small radiation doses to individuals and populations. The staff estimates of the expected liquid releases listed in Table 3.2 and the site hydrological considerations discussed in Sect. 2.3 of this statement and summarized in Table 5.6 were used to estimate radiation dose commitments to individuals and populations. The results of the calculations are discussed below.

Table 5.6. Summary of hydrologic transport and dispersion for liquid releases from SONGS 2 & 3[a]

Location	Transit time (hr)	Dilution factor
Nearest sport fishing location (plant outfall)[b]	0.1	1
Nearest shoreline (plant boundary)	0.1	1

[a]See Regulatory Guide 1.112, "Analytical Models for Estimating Radioisotope Concentrations in Different Water Bodies," (1976).

[b]Assumed for purposes of an upper-limit estimate; detailed information not available.

Radiation dose commitments to individuals

The estimated dose commitments to individuals at selected offsite locations where exposures are expected to be largest are listed in Tables 5.2 and 5.3. The standard NRC models given in Regulatory Guide 1.109 were used for these analyses.

Radiation dose commitments to populations

The estimated population radiation dose commitments to 80 km for SONGS 2 & 3 from liquid releases, based on the use of water and biota from the Pacific Ocean, are shown in Table 5.3. Dose commitments beyond 80 km were based on the assumptions discussed in Appendix B.

Background radiation doses are provided for comparison. The dose commitments from liquid releases from SONGS 2 & 3 represent small increases in the population dose from background radiation sources.

5.5.1.4 Direct radiation

Radiation from the facility

Radiation fields are produced in nuclear plant environs as a result of radioactivity contained within the reactor and its associated components. Doses from sources within the plant are

primarily due to nitrogen-16, a radionuclide produced in the reactor core. Since the primary coolant of pressurized water reactors is contained in a heavily shielded area of the plant, dose rates in the vicinity of PWRs are generally undetectable (less than 5 millirems per year). Low-level radioactivity storage containers outside the plant are estimated to contribute less than 0.01 millirem per year at the site boundary.

Occupational radiation exposure

The dose to nuclear plant workers varies from reactor to reactor and can be projected for environmental impact purposes by using the experience to date with modern pressurized water reactors (PWRs). Most of the dose to nuclear plant workers is due to external exposure to radiation from radioactive materials outside of the body rather than from internal exposure from inhaled or ingested radioactive materials. Recently licensed 1000 MWe PWRs are designed and operated in a manner consistent with the new (post-1975) regulatory requirements and guidelines. These new requirements and guidelines place increased emphasis on maintaining occupational exposure at nuclear power plants as low as is reasonably achievable (ALARA), and are outlined in 10 CFR Part 20, Standard Review Plan Chapter 12, and Regulatory Guide 8.8. The applicant's proposed implementation of these requirements and guidelines are reviewed by the NRC staff at the construction permit licensing stage, the operating license licensing stage, and during actual operation. Approval of the proposed implementation of these requirements and guidelines is granted only after the review indicates that an ALARA program can actually be implemented. As a result of our review the staff has determined that the applicant is committed to design features and operating practices that will assure that individual occupational radiation doses can be maintained within the limits of 10 CFR Part 20 and that individual and population doses will be as low as is reasonably achievable.

On the basis of actual operating experience, it has been observed that this occupational dose has varied considerably from plant to plant, and from year to year. Average individual and collective dose information is available from over 190 reactor-years of operation between 1974 and 1979. These data indicate that the average reactor annual dose at PWRs has been about 410 man-rem, with particular plants experiencing an average annual dose as high as 1300 man-rem. These dose averages are based on widely varying yearly doses at PWRs. For example, annual collective doses for PWRs have ranged from 18 to 5262 man-rem per reactor. The average annual dose per nuclear plant worker has been about 0.8 rem.

The wide range of annual doses (18 to 5262 man-rems) experienced by U.S. PWRs is dependent on a number of factors, such as the amount of required routine and special maintenance, and the degree of reactor operations and inplant surveillance. Since these factors can vary in an unpredictable manner, it is impossible to determine in advance a specific year-to-year or average annual occupational radiation dose for a particular plant over its operating lifetime. It is necessary to recognize that high doses may occur, even at plants with radiation protection programs that have been developed to assure that occupational radiation doses will be kept at levels that are ALARA. Consequently, the NRC staff's occupational dose estimates for environmental impact purposes for SONGS 2 and 3 are based on the conservation assumption that the station may have an higher than average level of special maintenance work. On the basis of the staff's review of the applicant's Safety Analysis Report, as well as occupational dose data from over 190 PWR reactor operating years, the NRC staff projects that the occupational doses at SONGS 2 and 3 could average as much as 1300 man-rem/yr when averaged over the life of the plant. However, actual year to year doses may differ greatly from this average, depending on actual plant operating conditions.

Transportation of radioactive material

The transportation of cold fuel to a reactor, of irradiated fuel from the reactor to a fuel reprocessing plant, and of solid radioactive wastes from the reactor to burial grounds is within the scope of the NRC report entitled "Environmental Survey of Transportation of Radioactive Materials to and from Nuclear Power Plants" [10 CFR 51.20(g)]. The estimated population dose commitments associated with transportation of fuels and wastes are listed in Tables 5.5 and 5.7.

5.5.1.5 Comparison of dose assessment models

The applicant's site and environmental data provided in the ER and in subsequent answers to staff questions were used extensively in the dose calculations. Any additional data received which could significantly affect the conclusions reached in this draft statement will be used in preparing the final statement.

Table 5.7. Environmental impact of transportation of fuel and waste to and from one light-water-cooled nuclear power reactor[a,b]

Exposed population	Estimated number of persons	Range of doses to exposed individuals (millirems per reactor year)[c]	Cumulative dose to exposed population (man-rems per reactor year)[d]
Transportation workers	200	0.01 to 300	4
General public			
Onlookers	1,100	0.003 to 1.3	
Along Route	600,000	0.001 to 0.06	3
Accidents in transport			
Radiological effects	Small[e]		
Common (nonradiological) causes	1 fatal injury in 100 reactor years; 1 nonfatal injury in 10 reactor years; $475 property damage per reactor year		

[a]Data supporting this table are given in the Commission's *Environmental Survey of Transportation of Radioactive Materials to and from Nuclear Power Plants,* WASH-1238, December 1972, and Suppl. I, NUREG-75/038, April 1975.

[b]Normal conditions of transport: heat (per irradiated fuel cask in transit), 250,000 Btu/hr; weight (governed by Federal or State restrictions), 73,000 lb per truck; 100 tons per cask per rail car; traffic density, <1 per day; rail <3 per month.

[c]The Federal Radiation Council has recommended that radiation doses from all sources of radiation other than natural background and medical exposures should be limited to 5000 milirems per year for individuals as a result of occupational exposure and should be limited to 500 millirems per year for individuals in the general population. The dose to individuals as a result of average natural background radiation is about 102 millirems per year.

[d]Man-rem is an expression for the summation of whole body doses to individuals in a group. Thus, if each member of a population group of 1000 people were to receive a dose of 0.001 rem (1 millirem), or if 2 people were to receive a dose of 0.5 rem (500 millirems) each, the total man-rem in each case would be 1 man-rem.

[e]Although the environmental risk of radiological effects stemming from transportation accidents is currently incapable of being numerically quantified, the risk remains small regardless of whether it is being applied to a single reactor or a multireactor site.

(To convert lb to kg, multiply by 0.45; to convert tons to tonnes, multiply by 0.907.)

5.5.1.6 Evaluation of radiological impact

The actual radiological impact associated with the operation of SONGS 2 & 3 will depend, in part, on the manner in which the radioactive waste treatment system is operated. The staff concludes on the basis of their evaluation of the potential performance of the radwaste system that the system as proposed is capable of meeting the dose design objectives of 10 CFR Part 50, Appendix I. Table 5.3 compares the calculated maximum individual doses to the dose design objectives. However, because the facility's operation will be governed by operating license technical specifications and because the technical specifications will be based on the dose design objectives of 10 CFR Part 50, Appendix I, as shown in the first column of Table 5.3, the actual radiological impact of plant operation may result in doses close to the dose design objectives. Even if this situation exists, however, the individual doses will still be very small when compared to natural background doses (∿100 millirems per year) or of the dose limits specified in 10 CFR Part 20. As a result the staff concludes that there will be no measurable radiological impact on man from routine operation of SONGS 2 & 3.

5.5.2 Radiological impacts to biota other than man

Depending on the pathway and the radiation source, terrestrial and aquatic biota will receive doses approximately the same or somewhat higher than man receives. Although guidelines have not been established for acceptable limits for radiation exposure to species other than man, it is generally agreed that the limits established for humans are also conservative for other species. Experience has shown that it is the maintenance of population stability that is crucial to the survival of a species, and species in most ecosystems suffer rather high mortality rates from natural causes. Although the existence of extremely sensitive biota is possible and increased radiosensitivity in organisms may result from environmental interactions with other stresses (e.g., heat, biocides, etc.), no biota have yet been discovered that show a sensitivity (in terms

of increased morbidity or mortality) to radiation exposures as low as those expected in the area surrounding SONGS 2 & 3. Furthermore, in all the plants for which an analysis of radiation exposure to biota other than man has been made, there have been no cases of exposures that can be considered significant in terms of harm to the species, or that approach the exposure limits to members of the public permitted by 10 CFR Part 20.[19] Since the BEIR Report[20] concluded that the evidence to date indicates that no other living organisms are very much more radiosensitive than man, no measurable radiological impact on populations of biota is expected as a result of the routine operation of this plant.

5.5.3 Environmental effects of the uranium fuel cycle

On March 14, 1977, the Commission presented in the *Federal Register* (42 FR 13803) an interim rule regarding the environmental considerations of the uranium fuel cycle. It was effective (by Amendment of September 12, 1978) through March 14, 1979 and revised Table S-3 of Paragraph (e) of 10 CFR Part 51.20.* In a subsequent announcement on April 14, 1978, (43 FR 15613), the Commission further amended Table S-3 to delete the numerical entry for the estimate of radon releases and to clarify that the table does not cover health effects. On July 27, 1979, the Commission approved a final rule setting out revised environmental impact values for the uranium fuel cycle to be included in environmental reports and environmental statements for reactors (44 FR 45362). The final rule reflects new and updated information relative to reprocessing of spent fuel and radioactive waste management as discussed in NUREG-0116, *Environmental Survey of the Reprocessing and Waste Management Portions of the LWR Fuel Cycle*,[21] and NUREG-0216,[22] which presents staff responses to comments on NUREG-0116. The rule also considers other environmental factors of the uranium fuel cycle, including aspects of mining and milling, isotopic enrichment, fuel fabrication, and management of low-and high-level wastes. These are described in the AEC report WASH-1248, *Environmental Survey of the Uranium Fuel Cycle*.[23]

Specific categories of natural resource use are included in Table S-3 of the final rule, which is reproduced in this statement as Table 5.8.† These categories relate to land use, water consumption and thermal effluents, radioactive releases, burial of transuranic and high- and low-level wastes, and radiation doses from transportation and occupational exposures. The contributions in Table 5.8 for reprocessing, waste management, and transportation of wastes are maximized for either of the two fuel cycles (uranium only and no recycle); that is, the cycle that results in the greater impact is used.

The following assessment of the environmental impacts of the fuel cycle as related to the operation of SONGS 2 & 3 is based on the values given in Table 5.8 and the staff's analysis of the radiological impact from radon releases. For the sake of consistency, the analysis of fuel-cycle impacts has been cast in terms of a model 1000 MWe LWR operating at an annual capacity factor of 80%. In the following review and evaluation of the environmental impacts of the fuel cycle, the staff conclusions would not be altered if the analysis were to be based on the net electrical power output of SONGS 2 & 3.

The total annual land requirement for the fuel cycle supporting a model 1000 MWe LWR is about 46 ha (114 acres). Approximately 5 ha (13 acres) per year are permanently committed land, and 40 ha (100 acres) per year are temporarily committed. (A "temporary" land commitment is a commitment for the life of the specific fuel-cycle plant, e.g., mill, enrichment plant, or succeeding plants. On abandonment or decommissioning, such land can be used for any purpose. "Permanent" commitments represent land that may not be released for use after plant shutdown and/or decommissioning.) Of the 40 ha per year of temporarily committed land, 32 ha (79 acres) are undisturbed and 9 ha (22 acres) are disturbed. Considering common classes of land use in the U.S.,‡ fuel-cycle land-use requirements to support the model 1000 MWe LWR do not represent a significant impact.

The principal water-use requirement for the fuel cycle supporting a model 1000 MWe LWR is that required to remove waste heat from the power stations supplying electrical energy to the enrichment step of this cycle. Of the total annual requirement of 43 x 10^6 m^3 (11,000 x 10^6 gal), about 42 x 10^6 m^3 are required for this purpose, assuming that these plants use once-through cooling. Other water uses involve the discharge to air (e.g., evaporation losses in process cooling) of about 0.6 x 10^6 m^3 per year and water discharged to ground (e.g., mine drainage) of about 0.5 x 10^6 m^3 per year.

*A notice of final rulemaking proceedings was given in the *Federal Register* of May 26, 1977 (42 FR 26987) that calls for additional public comment before adoption or final modification of the interim rule.

†A narrative explanation of Table 5.8 (Table S-3) was published in the *Federal Register* (46 FR 15154-75) on March 4, 1981.

‡A coal-fired power plant of 1000 MWe capacity using strip-mined coal requires the disturbance of about 81 ha (200 acres) per year for fuel alone.

Table 5.8. Summary of environmental considerations for uranium fuel cycle[a]

Normalized to model LWR annual fuel requirement (WASH-1248) or reference reactor year (NUREG-0116)

Natural resource use	Total	Maximum effect per annual fuel requirement or reference reactor year of model 1000-MWe LWR
Land, acres		
Temporarily committed[b]	100	
Undisturbed area	79	
Disturbed area	22	Equivalent to 110-MWe coal-fired power plant
Permanently committed	7.1	
Overburden moved, millions of metric tons	2.8	Equivalent to 95-MWe coal-fired power plant
Water, millions of gallons		
Discharged to air	160	Equals 2% of model 1000-MWe LWR with cooling tower
Discharged to water bodies	11,090	
Discharged to ground	127	
Total	11,377	Less than 4% of model 1000-MWe LWR with once-through cooling
Fossil fuel		
Electrical energy, thousands of megawatt hours	321	Less than 5% of model 1000-MWe LWR output
Equivalent coal, thousands of metric tons	117	Equivalent to the consumption of a 45-MWe coal-fired power plant
Natural gas, millions of standard cubic feet	135	Less than 0.3% of model 1000-MWe energy output
Effluents — chemical, metric tons		
Gases (including entrainment)[c]		
SO_x	4,400	
NO_x[d]	1,190	Equivalent to emissions from 45-MWe coal-fired power plant for a year
Hydrocarbons	14	
CO	29.6	
Particulates	1,154	
Other gases		
F^-	0.67	Principally from UF_6 production, enrichment, and reprocessing. Concentration within range of state standards — below level that has effects on human health
HCl	0.014	
Liquids		
SO_4^{2-}	9.9	From enrichment, fuel fabrication, and reprocessing steps. Components that constitute a potential for adverse environmental effect are present in dilute concentrations and receive additional dilution by receiving bodies of water to levels below permissible standards. The constituents that require dilution and the flow of dilution water are:
NO_3^-	25.8	
Fluoride	12.9	
Ca^{2+}	5.4	
Cl^-	8.5	NH_3 — 600 cfs
Na^+	12.1	NO_3 — 20 cfs
NH_3	10.0	*Fluoride — 70 cfs*
Fe	0.4	
Tailings solutions, thousands of metric tons	240	From mills only — no significant effluents to environment
Solids	91,000	Principally from mills — no significant effluents to environment
Effluents — radiological, curies		
Gases (including entrainment)		
Rn-222		Presently under reconsideration by the Commission
Ra-226	0.02	
Th-230	0.02	
Uranium	0.034	
Tritium, thousands	18.1	
C-14	24	
Kr-85, thousands	400	
Ru-106	0.14	Principally from fuel reprocessing plants
I-129	1.3	
I-131	0.83	
Tc-99	0.203	rresently under consideration by the Commission
Fission products and transuranics		
Liquids		
Uranium and daughters	2.1	Principally from milling — included in tailings liquor and returned to ground — no effluents; therefore, no effect on environment
Ra-226	0.0034	From UF_6 production
Th-230	0.0015	
Th-234	0.01	From fuel fabrication plants — concentration 10% of 10 CFR Part 20 for total processing 26 annual fuel requirements for model LWR
Fission and activation products	5.9×10^{-6}	
Solids (buried on site)		
Other than high level (shallow)	11,300	9100 Ci come from low-level reactor wastes and 1500 Ci come from reactor decontamination and decommissioning — buried at land burial facilities. Mills produce 600 Ci — included in tailings returned to ground; about 60 Ci come from conversion and spent-fuel storage. No significant effluent to the environment
TRU and HLW (deep)	1.1×10^7	Buried at Federal repository
Effluents — thermal, billions of British thermal units	4.063	Less than 4% of model 1000-MWe LWR
Transportation, person-rems	2.5	
Exposure of workers and general public		
Occupational exposure, person-rems	22.6	From reprocessing and waste management

[a]In some cases where no entry appears, it is clear from the background documents that the matter was addressed and that, in effect, this table should be read as if a specific zero entry had been made. However, there are other areas that are not addressed at all in this table. Table S-3 of WASH-1248 does not include health effects from the effluents described in this table or estimates of releases of Radon-222 from the uranium fuel cycle. These issues which are not addressed at all by this table may be the subject of litigation in individual licensing proceedings. Data supporting this table are given in the *Environmental Survey of the Uranium Fuel Cycle*, WASH-1248, April 1974; the *Environmental Survey of the Reprocessing and Waste Management Portions of the LWR Fuel Cycle*, NUREG-0116 (Suppl. 1 to WASH-1248); and the *Discussion of Comments Regarding the Environmental Survey of the Reprocessing and Waste Management Portions of the LWR Fuel Cycle*, NUREG-0216 (Suppl. 2 to WASH-1248). The contributions from reprocessing, waste management, and transportation of wastes are maximized for either of the two fuel cycles (uranium only and no-recyle). The contribution from transportation excludes transportation of coal fuel to a reactor and of irradiated fuel and radioactive wastes from a reactor which are considered in Table S-4 of Sect. 51.20(g). The contributions from the other steps of the fuel cycle are given in columns A – E of Table S-3A of WASH-1248.

[b]The contributions to temporarily committed land from reprocessing are not prorated over 30 years, because the complete temporary impact accrues regardless of whether the plant services 1 reactor for 1 year or 57 reactors for 30 years.

[c]Estimated effluents based on combustion of equivalent coal for power generation.

[d]1.2% from natural gas use and process.

On a thermal effluent basis, annual discharges from the nuclear fuel cycle are about 4% of those from the model 1000 MWe LWR using once-through cooling. The consumptive water use of 0.6×10^6 m^3 per year is about 2% of that of the model 1000 MWe LWR using cooling towers. The maximum consumptive water use (assuming that all plants supplying electrical energy to the nuclear fuel cycle used cooling towers) would be about 6% of that of the model 1000 MWe LWR using cooling towers. Under this condition, thermal effluents would be negligible. The staff finds that these combinations of thermal loadings and water consumption are acceptable relative to the water use and thermal discharges of the proposed project.

Electrical energy and process heat are required during various phases of the fuel-cycle process. The electrical energy is usually produced by the combustion of fossil fuel at conventional power plants. Electrical energy associated with the fuel cycle represents about 5% of the annual electrical power production of the model 1000 MWe LWR. Process heat is primarily generated by the combustion of natural gas. This gas consumption, if used to generate electricity, would be less than 0.3% of the electrical output from a 1000 MWe plant. The staff finds that the direct and indirect consumption of electrical energy for fuel-cycle operations are small and acceptable relative to the net power production of the proposed project.

The quantities of chemical, gaseous, and particulate effluents with fuel-cycle processes are given in Table 5.8. The principal species are SO_X, NO_X, and particulates. The staff finds, on the basis of data in a Council on Environmental Quality report,[24] that these emissions constitute an extremely small additional atmospheric loading in comparison with these emissions from the stationary fuel-combustion and transportation sectors in the U.S., i.e., about 0.02% of the annual national releases for each of these species. The staff believes such small increases in releases of these pollutants are acceptable.

Liquid chemical effluents produced in fuel-cycle processes are related to fuel-enrichment, -fabrication, and -reprocessing operations and may be released to receiving waters. These effluents are usually present in such dilute concentrations that only small amounts of dilution water are required to reach levels of concentration that are within established standards. Table 5.8 specifies the flow of dilution water required for specific constituents. Additionally, all liquid discharges into the navigable waters of the United States from plants associated with the fuel-cycle operations will be subject to requirements and limitations set forth in an NPDES permit issued by an appropriate state or Federal regulatory agency.

Tailings solutions and solids are generated during the milling process. These solutions and solids are not released in quantities sufficient to have a significant impact on the environment.

Radioactive effluents estimated to be released to the environment from reprocessing and waste management activities and certain other phases of the fuel-cycle process are set forth in Table 5.8. Using these data, the staff has calculated the 100-year involuntary environmental dose commitment* to the U.S. population. These calculations estimate that the overall involuntary total body gaseous dose commitment to the U.S. population from the fuel cycle (excluding reactor releases and the dose commitment due to radon-222) would be approximately 400 man-rems per year of operation of the model 1000 MWe LWR. The additional involuntary total body dose commitment to the U.S. population from radioactive liquid effluents due to all fuel-cycle operations other than reactor operation, estimated on the basis of the values given in Table 5.8, would be approximately 100 man-rems per year of operation. Thus, the estimated involuntary 100-year environmental dose commitment to the U.S. population from radioactive gaseous and liquid releases due to these portions of the fuel cycle is approximately 500 man-rems (whole body) per year of operation of the model 1000 MWe LWR.

At this time Table 5.8 does not address the radiological impacts associated with radon-222 releases. Principal radon releases occur during mining and milling operations and, following completion of mining and milling, as emissions from stabilized mill tailings and from unreclaimed open-pit mines. The staff has determined that releases from these operations for each year of operation of the model 1000 MWe LWR are as follows:

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*The environmental dose commitment (EDC) is the integrated population dose for 100 years; i.e., it represents the sum of the annual population doses for a total of 100 years. The population dose varies with time, and it is not practical to calculate this dose for every year.

Mining: (during active mining)[25]	4060 Ci
Mining: (unreclaimed open-pit mines)[26]	30 to 40 Ci/year
Milling and Tailings:[27] (during active milling)	780 Ci
Inactive Tailings:[27] (prior to stabilization)	350 Ci
Stabilized Tailings:[27] (several hundred years)	1 to 10 Ci/year
Stabilized Tailings:[27] (after several hundred years)	110 Ci/year

The staff has calculated population dose commitments for these sources of radon-222 using the RABGAD computer code described in Section IV.J of Appendix A of NUREG-0002.[28] The results of these calculations for mining and milling activities prior to tailings stabilization are shown in Table 5.9.

Table 5.9. Estimated 100-year environmental dose commitment per year of operation of the model 1000 MWe LWR

Radon-222 releases		Dose commitments (man-rems)		
Source	Amount (Ci)	Total body	Bone	Lung (bronchial epithelium)
Mining	4100	110	2800	2300
Milling and active tailings	1100	29	750	620
Total		140	3600	2900

When added to the 500 man-rem total body dose commitment for the balance of the fuel cycle, the overall estimated total body involuntary 100-year environmental dose commitment to the U.S. population from the fuel cycle for the model 1000 MWe LWR is approximately 600 man-rems. Over this period of time, this dose is equivalent to 0.00002% of the natural background dose of about 3,000,000,000 man-rems to the U.S. population.*

The staff has considered health effects associated with the releases of radon-222, including both the short-term effects of mining, milling, and active tailings and the potential long-term effects from unreclaimed open-pit mines and stabilized tailings. After completion of active mining, the staff has assumed that underground mines will be sealed, with the result that releases of radon-222 from them will return to background levels. For purposes of providing an upper-bound impact assessment, the staff has assumed that open-pit mines will be unreclaimed and has calculated that if all ore were produced from open-pit mines, releases from them would be 110 Ci/year of operation of the model 1000 MWe LWR. However, since the distribution of uranium ore reserves available by conventional mining methods is 66.8% underground and 33.2% open pit,[29] the staff has further assumed that uranium to fuel LWRs will be produced by conventional mining methods in these proportions. This means that long-term releases from unreclaimed open-pit mines will be 0.332 x 110 or 37 Ci/year of operation of the model LWR.

On the basis of these assumptions, the radon released from unreclaimed open-pit mines over 100- and 1000-year periods can be calculated to be about 3700 Ci and 37000 Ci/year of operation of the model reactor, respectively. The total dose commitments for a 100-1000-year period would be as follows:

*Based on an annual average natural background individual dose commitment of 100 mrem and a stabilized U.S. population of 300 million.

Time span	Total release	Population dose commitments (man-rems)		
		Total body	Bone	Lung (brochial epithelium)
100 years	3,700	96	2,500	2,000
500 years	19,000	480	13,000	11,000
1,000 years	37,000	960	25,000	20,000

The above dose commitments represent a worst-case situation since no mitigation circumstances are assumed. However, state and Federal laws currently require reclamation of strip and open-pit coal mines, and it is very probable that similar reclamation will be required for uranium open-pit mines. If so, long-term releases from such mines should approach background levels.

For long-term radon releases from stabilized tailings piles, the staff has assumed that these tailings would emit, per year of operation of the model 1000 MWe LWR, 1 Ci/year for 100 years, 10 Ci/year for the next 400 years, and 100 Ci/year for periods beyond 500 years. With these assumptions, the cumulative radon-222 release from stabilized tailings piles per operating year of the model reactor will be 100 Ci in 100 years, 4,090 Ci in 500 years, and 53,800 Ci in 1000 years[30]. The total body, bone, and bronchial epithelium dose commitments for these periods are as follows:

Time span	Total release	Population dose commitments (man-rems)		
		Total body	Bone	Lung (brochial epithelium)
100 years	100	2.6	68	56
500 years	4,090	110	2,800	2,300
1,000 years	53,800	1,400	37,000	30,000

Using risk estimators of 135, 6.9, and 22.2 cancer deaths per million man-rems for total body, bone, and lung exposures, respectively, the estimated risk of cancer mortality due to mining, milling, and active tailings emissions of radon-222 would be about 0.11 cancer fatalities per operating year of the model 1000 MWe LWR. When the risk due to radon-222 emissions from stabilized tailings over a 100-year release period is added, the estimated risk of cancer mortality over a 100-year period is unchanged. Similarly, a risk of about 1.2 cancer fatalities is estimated over a 1000-year release period per operating year of the model 1000 MWe LWR. When potential radon releases from reclaimed and unreclaimed open-pit mines are included, the overall risks of radon induced cancer fatalities per operating year of the model 1000 MWe LWR would range as follows:

0.11-0.19 fatalities for a 100-year period

0.19-0.57 fatalities for a 500-year period

1.2-2.0 fatalities for a 1000-year period

To illustrate: A single model 1000 MWe LWR operating at an 80% capacity factor for 30 years would be predicted to induce between 3.3 and 5.7 cancer fatalities in 100 years, 5.7 and 17 in 500 years, and 36 and 60 in 1000 years as a result of releases of radon-222.

These doses and predicted health effects have been compared with those that can be expected from natural-background emissions of radon-222. Using data from the National Council on Radiation Protection[31], the average radon 222 concentration in air in the contiguous United States is about 150 pCi/m^3, which the NCRP estimates will result in an annual dose to the bronchial epithelium of 450 mrem. For a stabilized future U.S. population of 300 million, this represents a total lung dose commitment of 135 million man-rems per year. Using the same risk estimator of 22.2 lung cancer fatalities per million man-lung-rems used to predict cancer fatalities for the model 1000 MWe LWR, estimated lung cancer fatalities alone from background radon-222 in the air can be calculated to be about 3000 per year or 300,000 to 3,000,000 lung cancer deaths over periods of 100 and 1,000 years respectively.

Other nuclides produced in the cycle, such as carbon-14, will contribute to population exposures in addition to the radon-related potential health effects from the fuel cycle. It is estimated that 0.08 to 0.12 additional cancer deaths may occur per operating year of the model 1000 MWe LWR (assuming that no cure or prevention of cancer is ever developed) over the next 100 to 1000 years, respectively, from exposures to these other nuclides.

These latter exposures can also be compared with those from naturally-occurring terrestrial and cosmic-ray sources, which average about 100 mrem. Therefore, for a stable future population of 300 million persons, the whole-body dose commitment would be about 30 million man-rems per year, or 3 billion man-rems and 30 billion man-rems for periods of 100 and 1000 years respectively. These dose commitments could produce about 400,000 and 4,000,000 cancer deaths during the same time periods. From the above analysis, the staff concludes that both the dose commitments and health effects of the uranium fuel cycle are insignificant when compared to dose commitments and potential health effects to the U.S. population resulting from all natural background sources.

5.6 SOCIOECONOMIC IMPACTS

5.6.1 Introduction

A 96-km (60-mile) radius of the San Onofre site circumscribes most of the metropolitan areas of Los Angeles and San Diego, the third and fourteenth largest cities, respectively, in the United States. Between 1970 and 1980, San Diego County had a 37.1% increase in population, reaching a total of 1,861,846 in 1980 and a density of about 170/km^2 (438/m^2).

Continued growth within 96 km (60 miles) of the San Onofre site is expected for the next three decades. The central portion of Orange County and the city of San Diego and its immediate environs are projected to be the major growth areas (ER, Sect. 2.1.3.2.2). The population growth rates within 16 km (10 miles) of the site are expected to fluctuate over the operating life of SONGS 2 and 3. The annual growth rate between 1976 and 1980 is expected to be 4.2%, decreasing to 0.3% between 1990 and 2000, and rising to 1.1% between 2010 and 2020 (ER, Sect. 2.1.3.1.1).

5.6.2 Impact of the construction labor force

A peak labor force of about 3000 workers was employed at SONGS 2 and 3 in late 1979. Of this number, the applicant has estimated that about 600 workers (20% of the peak labor force) have relocated to the southern California area (Sect. 2.2.3). Although the staff could not determine the exact location of these workers, current growth projections for the area indicate that the addition of 600 workers represents an insignificant impact. Between 1976 and 1980 the population in the area that is 16 to 80 km (10 to 50 miles) from the site was projected to increase 2.2% (ER, Sect. 2.1.3.2.1). The addition of 600 workers accounts for less than 0.1% of the growth expected during that time period.

Staff interviews with local and regional officials indicated that construction of SONGS 2 and 3 has had no impact on cities within 24 km (15 mi) of the site. Representatives of Southern California Association of Governments stated that it was doubtful that any significant impact attributable to plant construction could be identified in Orange County. The facts that (1) the majority of the work force commuted to site, (2) there was widespread busing to and from Orange County, Oceanside, Vista, Escondido, and San Diego, and (3) the region is currently experiencing rapid population growth support the staff's judgment that no significant social impact has occurred or is likely to occur due to in-migration of construction workers.

Cessation of large construction projects can result in varying degrees of economic dislocation to an area, especially if a previously underdeveloped commercial and service structure is expanded to meet the requirements of a large, short-term population influx. The southern California area has a well-developed infrastructure; thus, ending the construction phase of SONGS 2 and 3 is not expected to produce significant economic dislocation.

5.6.3 Impact of the operating labor force

The operation of SONGS 2 and 3 will employ about 200 workers. Table 5.10 provides an estimate for typical operating personnel requirements and types of employment positions at a two-unit pressurized-water reactor (PWR). The operations positions will be filled first by current members of I.B.E.W. Local No. 246. Positions unfilled will be offered to all Southern California Edison (SCE) employees, and if the position remains unfilled, SCE will advertise in local and regional newspapers (ER, p. S.2-175). Because of the diversified labor markets of Los Angeles and San Diego, the staff believes that at least 75% of these workers can be hired from within a 96-km (60-mile) radius of the site.

The applicant conducted surveys in March 1976 to determine the residential location of SONGS 1 workers. Seventy-five percent of these workers lived within 40 km (25 miles) of the San Onofre site, and 65% resided in Orange County, 30% in San Diego County, and 5% in Los Angeles and Riverside counties (ER, Appendix 8A, p. 10). The surveys further indicated that the cities of Carlsbad, Oceanside, San Clemente, San Juan Capistrano, and Vista were the major

Table 5.10. Operating personnel for a two-unit PWR

1 Plant superintendent	Warehouse staff
1 Assistant plant superintendent	1 Superintendent
2 Safety engineers	1 Assistant superintendent
	5 Clerks
Quality assurance staff	1 Truck driver
1 Superintendent	Engineering section
4 Engineers	1 Superintendent
6 Engineering aides	3 Instrument engineers
	3 Instrument engineering aides
Administrative services	2 Senior instrument mechanic foremen
1 Superintendent	20 Mechanics
1 Assistant superintendent	2 Mechanical engineers
3 Payroll clerks	3 Mechanical engineering aides
9 Stenographers and file clerks	1 Reactor engineer
7 Janitors	1 Reactor engineering aide
	2 Nuclear engineers
1 Industrial engineer	1 Chemical engineer
1 Nurse	9 Chemical engineering aides
Health physics staff	Maintenance staff
1 Superintendent	1 Superintendent
2 Technicians	1 Assistant superintendent (electrical)
1 Clerk	1 Assistant superintendent (mechanical)
	2 Mechanical maintenance engineers
Security staff	1 Electrical maintenance engineer
1 Superintendent	3 Engineering aides
1 Assistant superintendent	
9 Security officers	Trades and labor staff
	1 Machinist foreman
Operations	11 Machinists
Control room staff	1 Boiler-maker foreman
1 Superintendent	5 Boiler makers
1 Assistant superintendent	1 Steam-fitter foreman
1 Training coordinator	12 Steam fitters
5 Clerks	1 Electrician foreman
6 Shift engineers	10 Electricians
10 Assistant shift engineers	1 Labor foreman
15 Unit operators	10 Laborers
18 Assistant unit operators	2 Truck drivers
	2 Carpenters
Communications engineering staff	2 Sheet metal workers
2 Engineers	2 Painters
3 Engineering aides	2 Insulators
	1 Structural iron worker

Source: Tennessee Valley Authority, Department of Planning, Chattanooga, Tenn., 1977.

communities of worker residence. The staff estimates that approximately the same pattern of location will occur with SONGS 2 and 3 workers as occurred with SONGS 1 workers.

Between 1973 and 1980, northern San Diego County was expected to have a population increase of about 22,000. From 1975 to 1980 southern Orange County was projected to grow by about 21,000 persons. Assuming that all operations workers relocated to the area, the staff concludes that the addition of 200 workers and their households represents a negligible effect.

The staff cannot determine precisely the number of workers who will (1) relocate from outside the area or (2) choose to move from within the 96-km (60-mile) radius to a residence closer to the plant. In order to predict the maximum possible impact on housing in the area, the staff assumes that all of the workers will relocate and thus require housing. A relocating operations force will likely demand permanent housing. From Table 5.11, it appears that housing availability in Orange and San Diego counties is sufficient to provide diversity in location for all operations workers' households. The table further indicates that, based on the number of vacant units in 1976, a surplus of housing exists in each of the communities expected to house workers.

Estimates on the location of SONGS 1 worker indicate SONGS 2 and 3 households will likely contribute to increased enrollments in the school districts of Carlsbad, Capistrano, Oceanside, Saddleback Valley, and Vista. The total additional enrollment at all five school districts

Table 5.11. Housing availability in Orange and San Diego counties

Communities	Residential distribution of households SONGS 2 & 3	Number of existing dwelling units as of Jan. 1, 1976	Number of vacant units as indicated by number of idle electric meters for Jan. 1, 1976
Orange County total	127	592,932	10,080
San Clemente	32	10,636	170
San Juan Capistrano	41	4,561	73
Saddleback (Irvine)	22	11,102	178
Other unincorporated areas	32	76,260	1,220
San Diego County total	61	547,708	8,763
Carlsbad	11	9,111	200
Oceanside	25	20,835	458
Vista	20	12,539	276
Other unincorporated areas	5	108,841	2,395

Source: ER, Suppl. 2, Table 89-A, p. S.2-178.

will be about 105 students (ER, Appendix A, p. 20). The community college districts of Oceanside-Carlsbad, Palomar, and Saddleback will likely increase their enrollments by approximately 20 to 25 students (ER, Appendix 8A, p. 20). The staff concludes that this estimated increased enrollment represents a negligible impact on the school districts.

Operations employment at SONGS 2 and 3 will be relatively high-paying, stable work. About 87% of the total work force will have gross incomes in excess of $15,000 per year (ER, Appendix 8A, p. 15). The annual average income in 1976 dollars for a SONGS 2 and 3 household will be about $20,800. This compares to a median family income in 1980 for San Diego and Orange counties of $21,500 and $26,200 respectively. SONGS 2 and 3 households are expected to contribute to the economic activity of the area. Total taxable retail expenditures by households of operations employees are estimated to be about $855,000 per year (ER, p. S.2-176). In addition, those workers who build homes will contribute further to the economic activity of the area.

5.6.4 Economic impacts

The staff believes that the major economic impact associated with the operation of SONGS 2 and 3 will be a result of tax revenues generated by the plant. These taxes include property tax, state income tax, utility users tax, franchise tax payments, and sales and use taxes. The analysis presented here differs from that presented earlier in the DES by taking into account the impacts of the Jarvis-Gann Amendment (Proposition 13). The following discussion is based on two important assumptions. (1) The method of determining the value of state-regulated utility systems, currently before the State Court of Appeals, will be decided in accordance with the decision of the State Board of Equalization. Accordingly, SONGS 2 and 3 will be assessed on current market value, based on historical methods of valuation rather than on the 1975-76 base year as prescribed in Proposition 13. (2) The allocation of tax revenues among the various funds and districts within the county will remain roughly the same as at present.[32] Changes in either of the above conditions in the future may result in significant variation from the situation described here.

Under Proposition 13, neither the assessed value of the SONGS 2 and 3 units nor their annual tax liability differs greatly from the figures presented in the DES. Earlier projections were for an assessed valuation of $348 million in 1976 dollars (ER, Appendix 8-A, p. 4) and an annual property tax payment of $13.1 million (DES, Sect. 5.6.4). Current calculations show an eventual assessed value of $326 million in 1979 dollars and an annual tax of approximately $13 million (Table 5.12). At present, current construction at SONGS 2 and 3 is already assessed at roughly $100 million and is generating $4 million yearly in property tax revenues. The remaining $9 million in property taxes will be added as construction is completed.[32]

While the total tax burden is not significantly different under the terms of Proposition 13, the distribution of the resulting revenues is. Previously, it was projected that nearly all of the $13 million in property taxes generated by SONGS 2 and 3 would go to the County General Fund, the County Library Fund, and three local school districts in the immediate vicinity of the plant – Fallbrook Union Elementary, Fallbrook Union High, and Palomar Community College (DES,

Table 5.12. Projected impacts of SONGS 2 & 3 on San Diego County property tax revenues

	San Diego County	SONGS 2 & 3	Total: County plus SONGS 2 & 3	SONGS 2 & 3 as % of total
Assessed value	$7,775.5 million[1]	$326 million[3]	$8,101.5 million	4.0%
Annual taxes	$311 million[2]	$13 million[3]	$324 million	4.0%

[1]For FY 1978-79, not counting $100 million of SONGS 2 & 3 construction currently on tax rolls.
[2]For FY 1978-79, not counting $4 million currently received for SONGS 2 & 3 construction.
[3]As of project completion, in 1979 dollars.
Source: Letter from J. H. Drake, Southern California Edison Co., to W. H. Regan, Jr., U.S. NRC, dated April 17, 1979.

Sect. 5.6.4). Now, however, the new revenues will be distributed throughout the county on the basis of the historical property tax revenue relationships between all the various funds and districts. Accordingly, the five entities named above will receive roughly one-fourth of the plant-induced taxes, or $3.4 million, because this is the proportion of all county funds they have traditionally received. The remaining $9.6 million will go to other recipients county-wide. Because of this widespread distribution, the property taxes paid by SONGS 2 and 3 will not bring a large windfall to any single district but, rather, a modest 4.0% increase to all county funds and districts over pre-construction receipts (2.9% over the present situation where $100 million of plant construction is already on the tax rolls). The debt service rate of the three previously named school districts will be reduced as a result of plant induced revenues but this represents a very small part of the total property tax.[32]

Sales and use taxes payable to the State of California are levied at 6% of the retail or use value of fixtures, equipment, machinery, and materials purchased either in or outside of the State of California and placed in use within the state. For every 6 cents collected, 1.25 cents is allocated to counties and cities. The state tax on nuclear fuel for SONGS 2 and 3 is expected to be about $2.5 million per year. In addition, $415,000 in sales tax for materials will be paid in 1981, the first year of operation (ER, Appendix 8A, p. 8).

Over the operating life of SONGS 2 & 3, about $66 million in California state corporate income taxes will be paid by the applicant. California also has a City Utility Users Tax that, although it is difficult to determine the proportion for which SONGS 2 & 3 are directly responsible, is estimated to increase by $1.6 million per year (ER, Appendix 8A, p. 8). This tax varies for each city, and the revenues are not earmarked for any particular purposes.

The California Energy Resources Surcharge is included in the retail customer's bill and is collected by the utility. The current surcharge is $0.00015 per kilowatt-hour. The revenues collected are placed in the State Energy Resources Conservation and Development Special Account in the General Fund in the State Treasury by the State Board of Equalization. All funds in the account are to be expended for the purpose of carrying out the provisions of the Warren-Alquist State Energy Resources Conservation and Development Act.

5.6.5 Impact on recreational resources

In the early 1960s the applicant secured a leasehold from the U.S. Marine Corps at Camp Pendleton. During construction of SONGS 1, the Marine Corps released about 5.6 km (3.5 miles) of beach front to the State of California to be maintained as San Onofre State Beach. When this park opened in 1971, an additional 2440 m (8000 ft) of beach front had gained public access. Of this, 1370 m (4500 ft) are on the applicant's leasehold and the remaining 1070 m (3500 ft) are immediately north of the plant site, comprising another section of the state beach.

In order to comply with NRC regulations regarding the siting of nuclear power plants set forth in 10 CFR Part 100, the applicant proposes to control recreational activities on the beach for a distance of about 1.4 km (0.85 mile) adjacent to the station (ER, Sect. 2.1.2). Access to this area will be permitted for the purpose of viewing the barrancas and bluffs south of the station and for pedestrian passage between the public beach areas north and south of the station. Recreational activities, such as sunbathing or picnicking, will not be permitted within the landward portion of this restricted area. To facilitate passage between the beaches, a walkway will be constructed through the restricted area adjacent to the seawall. This walkway will be 4.6 m (15 ft) wide, will be bounded by a 2.4-m (8-ft) chain link fence, and will be used only for passage through the restricted area. It is the judgment of the staff that the fence proposed by the applicant is inappropriate in light of the scenic nature of the area and that a less aesthetically objectionable way should be

sought to restrict access to the beach. Therefore, it is recommended that the applicant consider alternate methods of beach enclosure that will safely restrict access in a manner compatible with the scenic nature of this area.

In the Final Environmental Statement required for the construction permit of SONGS 2 and 3, the staff stated, "Use of the beach will not be restricted after construction is complete" (FES-CP, p. 2-11). The current plan to restrict use of approximately 1.4 km (0.85 mile) of the beach front for the 30-year operating life of the plant is a significant loss of valuable recreational and scenic space and represents a substantial change in action between issuance of the FES-CP and application for an operating license. The staff further stated, "The beach in the vicinity of the Station (5639 m (18,500 ft) south and 1036 m (3400 ft) north) is considered to be a unique and scarce recreational resource," (FES-CP, p. 2-11) and "Closure for even a brief period is objectionable" (FES-CP, p. 8-1). The loss of this resource precludes recreational benefits to significant numbers of beach users in the vicinity of San Onofre Beach. The staff reiterates those judgments and concludes that the current plan to restrict the public's use of this beach is a significant cost of the project, unanticipated at issuance of the construction permit. This impact is not sufficiently adverse, however, to warrant denying an operating license.

While all state beaches in the Pendleton coast area experienced increased usage in recent years, the attendance at San Onofre State Beach has risen significantly faster than at the other facilities. Between 1972 and 1978, the annual number of visits to the San Onofre State Beach rose by 98% while San Clemente and Doheny State Beaches showed increases of 46% and 25%, respectively (ER, Appendix 8A, Table 24, and Reference 32). As demand on available recreational resources increases, the significance of removing the beach in front of SONGS 2 & 3 from unrestricted public use will increase.

5.6.6 Emergency planning impacts

The applicants are currently revising the Emergency Plan, San Onofre Nuclear Generating Station Units 2 and 3 in accordance with 10 CFR Part 50, as amended July 23, 1980, as well as the recommended criteria contained in NUREG-0654. The staff believes the only noteworthy potential source of impact on the public from emergency planning would be associated with the siren alert system. The system will be designed to provide a minimum 10db dissonant differential from the ambient noise levels. The maximum sound level received by any member of the public should be lower than 123db. A complete cycle test will be required annually. The test requirements and alarm noise levels are consistant with those used for existing alert systems; therefore the staff concludes that the noise impacts associated with the siren alert system will be infrequent and insignificant.

5.6.7 Summary and conclusion

The staff concludes that, with the significant exception of restricting public use of 1.4 km (0.85 mi) of the San Onofre beach, the social and economic impact of operating SONGS 2 & 3 will be moderate. The large population within 96 km (60 miles) of the site and the projected population growth in the area is such that the addition of all 200 workers and their families would represent negligible impact to the area. Under the terms of Proposition 13, the property tax revenues received by the various funds and districts in San Diego County will be relatively small in proportion to existing revenues.

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* Available for inspection and copying for a fee in the NRC Public Document Room, 1717 H St., NW., Washington, DC 20555.
** Available from the NRC/GPO Sales Program, Washington, DC 20555, or the National Technical Information Service, Springfield, VA 22161.
***Available from NTIS only.

6. ENVIRONMENTAL MONITORING

6.1 SUMMARY

The applicant has expanded its San Onofre Unit 1 environmental monitoring program (biological, chemical, physical, and thermal) to determine environmental effects which may occur as a result of site preparation and construction of Units 2 and 3 and to establish an adequate preoperational baseline by which the operational effects of Units 2 and 3 may be judged.

The aquatic preoperational environmental monitoring program for SONGS 2 and 3 was approved by NRC and implemented by the applicant in April 1978. The NRC-approved program terminated in September 1980. However, all NPDES permit monitoring program requirements will continue to be met until an approved operational monitoring program is implemented. Results of the preoperational monitoring program will be used in formulating the operational monitoring program, which the applicant will submit for approval by the California Regional Water Quality Control Board to be incorporated in the NPDES permit monitoring program.

The environmental monitoring programs presented here differ somewhat from the description in the FES-CP. More detailed information is given here than in the FES-CP. Two state agencies, the California Regional Water Quality Control Board and the California Coastal Commission, have imposed environmental monitoring requirements in the vicinity of the San Onofre Station. NRC has discussed the results of its environmental review with the State agencies and has provided the State with recommendations for monitoring. The sections which follow include NRC staff recommendations based on its environmental review. However, requirements for non-radiological monitoring of the aquatic environment will be the responsibility of the State.

6.2 PREOPERATIONAL ENVIRONMENTAL PROGRAMS

The results from the preoperational monitoring program for Units 2 and 3 will be submitted with the Annual Operating Report for Unit 1.

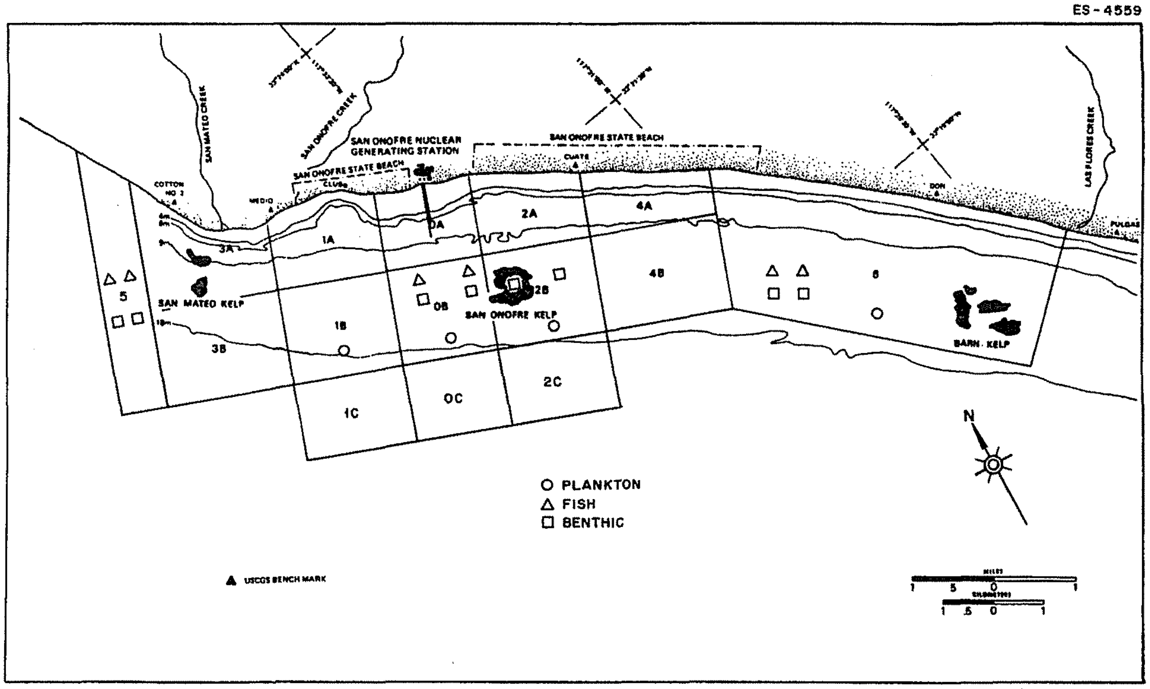
6.2.1 Aquatic biological monitoring program

The applicant's preoperational aquatic biological monitoring program was designed to determine the species composition, abundance, and the temporal and spatial distribution of phytoplankton, zooplankton, ichthyoplankton, nekton, benthos, kelp beds, and intertidal organisms. The data obtained will be used to provide a basis for comparison with future operational monitoring data to determine if plant operation has caused observable perturbations in the ecosystem.

The possible operational impacts identified in this document and the FES-CP include: changes in local plankton populations due to entrainment; changes in the abundance of fish eggs, larvae, juveniles, and adults due to entrainment; adult fish population shifts due to fish impingement; alterations in some of the benthic and fish communities from thermal discharges; and changes in benthic and planktonic communities from increased turbidity. Thus, results from the preoperational and operational monitoring programs will be used to determine the extent to which the above effects occur.

6.2.1.1 Phytoplankton and zooplankton

Phytoplankton and zooplankton were sampled bimonthly. Samples were collected from at least four fixed stations, one each in zones 0B, 1B, 2B and 6 (Figure 6.1). A pump system is used to sample the water column and a 202 μm mesh-size screen is used to collect the zooplankton. Zooplankton biomass is determined and predominant species are enumerated. Chlorophyll analyses are performed on whole-water samples. Collections are coordinated, as much as possible, with the collection of pertinent physical data such as temperature, transparency, and current velocity and direction.



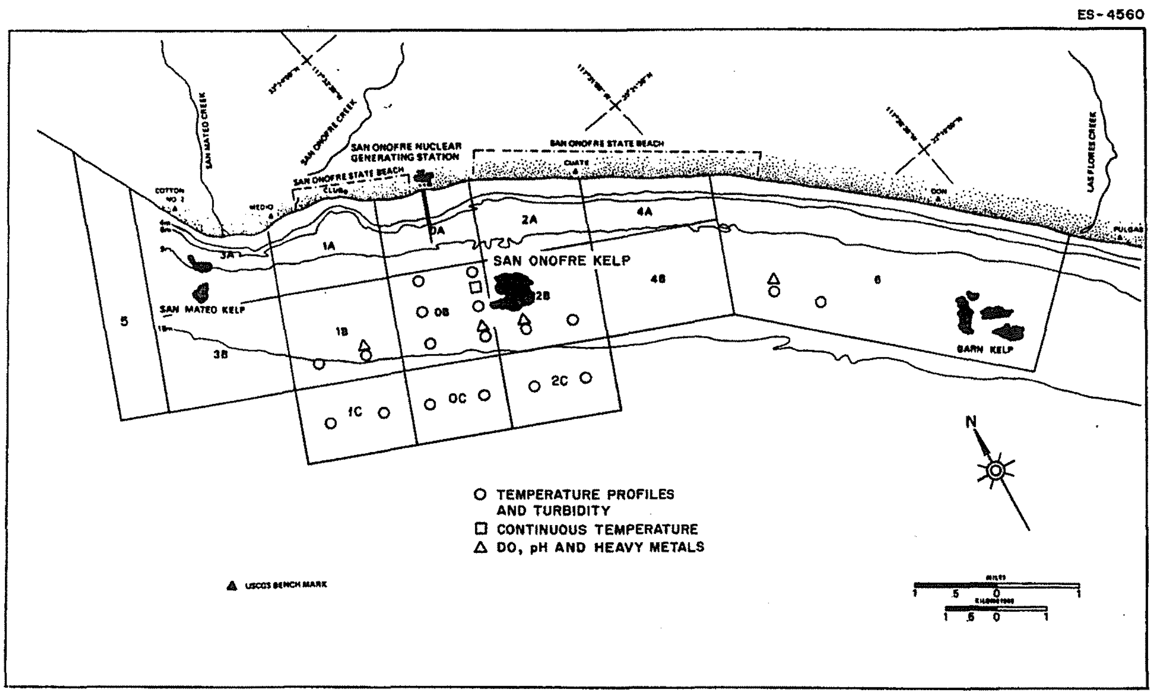


Fig. 6.1. Environmental monitoring stations for SONGS 2 and 3 preoperational monitoring program. Source: ER, Apendix 6A, Figs. 1 and 2.

The staff recommends that predominant phytoplankton genera also be enumerated to provide baseline conditions for this group. This would enable, for example, the determination of whether operation of the facility promotes red tide development (see Sect. 5.3.2, FES-CP).

6.2.1.2 Ichthyoplankton

Ichthyoplankton will be collected monthly at two stations in the Units 2 and 3 discharge area, zone OB, and at two stations in the reference area, either zone 5 or 6. Additionally, the Unit 1 intake area will be sampled. The study began approximately two years prior to initial operation of Unit 2 and lasted one year. Sampling was conducted during the day, at night, at dawn, and at dusk at the intake; night sampling was employed at the other locations. The water surface, water column, and epibenthos was sampled at each station. Fish larvae were identified to the lowest taxon possible and enumerated. Fish eggs were sorted and enumerated.

A study by the Marine Review Committee (MRC) was initiated in July 1976 (see Section 6.4.2) to assess the distribution, abundance, and entrainment of ichthyoplankton at SONGS 1. It is expected that data acquired from this work will also help characterize the SONGS 2 and 3 environment.

6.2.1.3 Nekton

Replicate fish samples were collected on a quarterly basis from at least two stations in zone OB, two in zone 5, two in the control zone, zone 6 (Figure 6.1). The gill nets used were 2- by 46-m (6- by 150-ft) full size, containing six 7.5-m (25-ft) panels of 19.05-, 25.4-, 31.75-, 38.1-, 44.5-, and 63.5-mm (3/4-, 1-, 1-1/4-, 1-1/2-, 1-3/4-, and 2-1/2-in.) bar mesh. The fish were measured, their state of health was assessed, and sexual maturation was determined on subsamples. Synoptic measurements of temperature and transmissivity were taken at each station.

6.2.1.4 Benthos

Benthic samples were collected quarterly at at least two stations within each of zones OB, 2B, 6 and 5 (or zones 3A and/or 3B) (Fig. 6.1). Permanent sampling stations exist in which a 6-m^2 (64.56-ft^2) sampling area has been established. Each sampling area contains 300 evenly spaced contact points which are used to estimate the distribution and relative abundance of sessile invertebrates, large motile invertebrates and macrophytes. Species enumeration and substrate type are recorded for each contact point. Additionally, four 0.125 m^2 (1.35-ft^2) quadrants are randomly placed within the sampling area to evaluate the distribution and abundance of small, clumped, or patchily distributed organisms. General observations to be recorded during sampling include: quantity and composition of drift algae, conspicuous or sparsely distributed biota not sampled with the point contact method, and substrate alteration (e.g., increased sedimentation). Selected species which are enumerated will be measured, and their general condition recorded. Procurement of some of the physical data, such as temperature and turbidity, will be coordinated with the benthic sampling program.

6.2.1.5 Intertidal organisms

Although not a required component of the preoperational monitoring program, quarterly observations were made along cobble intertidal transects at four monitoring stations and one control station. Predominant macroscopic species and substrate composition were identified and enumerated within three permanent 0.25-m^2 (2.69-ft^2) quadrats along a line perpendicular to the beach. Photographs were also taken of each quadrat for a permanent record of any possible ecological changes.

The staff believes that it is unnecessary to begin the intertidal sampling program until the time of removal of the construction apron from SONGS 2 and 3 (See FES-CP, Sect. 4.3.2, p. 4-9). At that time the intertidal monitoring program should be reinstated to assess the effect of the added sand movement in the intertidal zone. Provided the data show no significant effects, this program may be terminated after all translocation of sand has occurred or after two years. Until the time of apron removal, visual inspection of the intertidal zone will be sufficient, with biological sampling and laboratory analysis initiated only if needed. Deletion of the intertidal program may be reasonable during operational monitoring because of the extensive impact sustained by the intertidal area from activities unassociated with SONGS (Sect. 2.5.2.4) and because of the unlikely potential for any significant impact resulting from SONGS operation.

6.2.1.6 Kelp beds

The three kelp beds, San Mateo, San Onofre, and Barn, located near SONGS (Fig. 6.1) are being studied. A brief outline of the scope of effort at the three kelp beds is as follows:

1. Three benthic stations are located in and about the San Onofre kelp bed and one each at Barn kelp and San Mateo kelp. Stations are quantitatively assessed quarterly.

2. Kelp canopies and rock substrate are mapped for areal extent on a quarterly basis.

3. Water nutrient analysis for ammonia, nitrates, nitrites, and phosphate are taken monthly at all three beds. Water samples are taken from the surface and bottom from within each bed and offshore of each bed. An additional offshore station serves as a monitoring area for upwelling.

4. Kelp tissue analysis for nutrient content is conducted on a monthly basis at all three kelp beds. Each leaf is analyzed for nitrogen content.

5. An assessment of the health of the kelp plants in the three beds is made on a quarterly basis. Parameters assessed include: success of juvenile recruitment, density of kelp plants, amount of encrusting organisms and grazing by herbivores and abundance of senile and diseased plants.

6. Aerial infra-red photographs of the three kelp bed canopies will be taken on a monthly basis.

6.2.2 Water quality monitoring program

The preoperational water quality monitoring program is an expansion of the existing program required by the Environmental Technical Specifications for SONGS 1. This program is designed to establish baseline characteristics of selected oceanographic parameters for comparison with data obtained during the operation of SONGS. This comparison will allow determination of the extent to which SONGS operation alters water quality. Those parameters identified in the FES-CP and in this document which might be altered include: pH, temperature, turbidity, certain heavy metals, and dissolved oxygen.

Sea water temperature-depth profiles are measured bimonthly at stations in the area of the Units 2 and 3 diffusers and at a reference station outside of the area of predicted thermal influence. Stations are as follows: two within each of zones 1B, 2B, 1C, OC, 2C, and 6, six stations within zone OB (Fig. 6.1). Additionally, sea water temperatures are continuously monitored near the surface, at mid-depth, and near the bottom at a permanent station in zone OB. Temperatures from each depth are recorded hourly. The accuracy of the system is ±0.5 degrees centigrade, ±30 minutes per month.

Turbidity is monitored bimonthly at two stations within each of zones 1B, 2B, 1C, OC, 2C, and 6, and at six stations within zone OB (Fig. 6.1). The pH is monitored bimonthly at four sampling stations — one in each of zones OB, 1B, 2B, and 6. Dissolved oxygen is measured bimonthly at four stations — one in each of zones OB, 1B, 2B, and 6.

Mid-depth ocean water samples and grab samples of ocean bottom sediments are collected quarterly in the area of the Units 2 and 3 diffusers and an appropriate control area for analysis of heavy metals. One station in each of zones 1B, 2B, OB, and 6 is sampled. Samples will be analyzed for chromium, iron, and titanium. Copper will not be monitored as the applicant has indicated that SONGS 2 and 3 will have titanium condenser tubing.

The staff considers this program adequate with the following additions: (1) the water quality data should be collected within a two-day period at maximum to permit station-by-station comparisons and the investigation of possible cause and effect relationships, and (2) all control samples should be collected from an area predicted to be unaffected by any discharge effect.

6.2.3 Terrestrial monitoring program

The baseline terrestrial environmental monitoring program for the FES-CP was very nominal. As a condition of the construction permit, the applicant expanded its terrestrial monitoring program to establish an adequate preoperational baseline by which the operational effects of SONGS 2 and 3 may be judged. Biological data were collected seasonally in order to document changes in the biotic communities over a one-year time span. Methods utilized included

small mammal trapping; bird censusing; observations of reptiles, amphibians, and large mammals; plant species lists; and vegetation analyses using the line intercept and quadrat methods. Results of this expanded monitoring program are presented in Sect. 2.5.1.

The applicant has proposed and is currently monitoring areas of cut and fill associated with construction of the plant and transmission lines to detect areas of erosion (ER, Appendix 6A, Special Studies I). Visual inspections are conducted and documented biweekly; any erosion resulting from the applicant's construction activities will receive appropriate corrective action.

6.2.4 Radiological monitoring program

Radiological environmental monitoring programs are established to provide data on measurable levels of radiation and radioactive materials in the site environs. Appendix I to 10 CFR Part 50 requires that the relationship between quantities of radioactive material released in effluents during normal operation, including anticipated operational occurrences, and resultant radioactive doses to individuals from principal pathways of exposure be evaluated. Monitoring programs are conducted to verify the effectiveness of in-plant controls used for reducing the release of radioactive materials and to provide public reassurance that undetected radioactivity will not build up in the environment. A surveillance program is established to identify changes in the use of unrestricted areas to provide a basis for modifications of the monitoring programs.

The preoperational phase of the monitoring program provides for the measurement of background levels and their variations along the anticipated important pathways in the area surrounding the plant; the training of personnel; and the evaluation of procedures, equipment, and techniques.

This is discussed in greater detail in NRC Regulatory Guide 4.1, Rev. 1, "Programs for Monitoring Radioactivity in the Environs of Nuclear Power Plants," and the Radiological Assessment Branch Technical Position, August 1977, "Standard Technical Specification for Radiological Environmental Monitoring Program."

The applicant has proposed a radiological environmental monitoring program to meet the objectives discussed above. The applicant's proposed preoperational radiological environmental monitoring program is presented in Sect. 6.1.5 of the applicant's Environmental Report.

The applicant proposes to initiate parts of the program two years prior to operation of the facility, with the remaining portions beginning either six months or one year prior to operation.

The staff concludes that the radiological preoperational monitoring program proposed by the applicant is acceptable.

6.2.5 Onsite meteorological monitoring program[1,2,3]

The original onsite meteorological program began in late 1964 with wind measurements at the top of a 19.5-m (64-ft) mast. In December 1970, the current meteorological monitoring program began with the installation of a 36.6-m (120-ft) tower atop the coastal bluff about 100 m (330 ft) west-northwest from the Unit 1 containment and 420 m (1380 ft) west-northwest of the Unit 2 containment. In October 1975 the tower was extended to a height of about 43 m (140 ft). Table 6.1 describes the kinds of measurements and their elevations on the tower between 1970 and the present.

Southern California Edison Company also conducted an onshore tracer test program at the San Onofre site. Among the objectives of the program were (1) to evaluate the appropriateness of using data measured on the existing site meteorological tower located on the coastal bluff for making dispersion estimates for onshore flows, and (2) to characterize dispersion representative of meteorological conditions during routine plant releases. NUS-1927[3] describes the test program and data.

On the basis of our analysis of the test data, we conclude that the wind and vertical temperature data measured on the San Onofre onsite (bluff) tower are acceptable for use in calculating atmospheric dispersion estimates for the site vicinity using the staff's model, described in Sect. 2.4.4.

Table 6.1. SONGS onsite meteorological instrumentation

Period	Measured parameter	Elevation above ground	
		Meters	Feet
December 1970–January 1973	Wind direction, speed and standard deviation	36.6	120
	Dry bulb vertical temperature gradient	36.6–6.1	120–20
January 1973–October 1975	Wind direction and speed	10, 36.6	33, 120
	Wind direction standard deviation	36.6	120
	Dry bulb temperature[a]	6.1	20
	Wet bulb temperature[b]	6.1	20
	Dry bulb vertical gradient	36.6–6.1	120–20
October 1975–present	Wind direction and speed	10, 20,[c] 40	33, 66, 131
	Wind direction standard deviation	10	33
	Dry bulb temperature	10	33,
	Dry bulb vertical gradient	40–10[d] 36.6–6.1[c]	131–33 120–20

[a]Installed January 1974.
[b]Installed January 1974, removed January 1975.
[c]Temporary.
[d]Two sets of instruments.

6.3 OPERATIONAL MONITORING PROGRAMS

6.3.1 Aquatic biological monitoring program

The aquatic biological operational monitoring program will contain sampling programs which are extensions of the baseline and preoperational programs so that analyses can readily be made of the changes, if any, that occur in the aquatic environment due to plant operation. Thus, the ichthyoplankton study now being conducted and the required kelp preoperational program should be continued during operation of the facility until such time as it is possible to state credibly that no significant impacts result from the facility.

The new fish return system (Sect. 3.2.2) is expected to be about 90% effective according to laboratory models (ER, p. 5.1-20). Precise figures on its effectiveness will not be available until it is operated in conjunction with the heat dissipation system. The staff recommends that the applicant include a program for optimizing the effectiveness of the fish return system. This should include consideration of the delayed mortality of the fish successfully diverted by the fish return system by holding them for 48 to 96 hours before returning them to the ocean.

Consideration of deletion of the intertidal sampling program from the operational monitoring program for SONGS 2 and 3 is discussed in Sect. 6.2.1.5.

6.3.2 Water quality monitoring program

The water quality operational monitoring program is a continuation of the existing preoperational water quality monitoring program (Sect. 6.2.2). This continuity will allow for confirmatory monitoring to assess any possible changes to water quality due to operation of San Onofre Units 2 and 3.

The NRC and the California Regional Water Quality Control Board, San Diego Region (CRWQCB) have worked in a cooperative manner in order to develop the preoperational monitoring program for SONGS 2 and 3. NRC and CRWQCB have agreed to continue to work together to establish an operational phase NPDES permit which will incorporate the aquatic concerns from each regulatory group.

6.3.3 Terrestrial monitoring program

The applicant does not have an operational terrestrial monitoring program. The staff does not recommend any operational monitoring of floral or faunal species because no significant

effects have been identified between the operation of SONGS 2 & 3 and the terrestrial environment. The California Coastal Commission, however, requires the applicant to protect the bluffs 0.5 km (0.31 mile) south of the plant site for the duration of the site easement (expiration date, May 1, 2023) (ER, Appendix 12B).

6.3.4 Radiological monitoring program

The operational offsite radiological monitoring program is conducted to measure radiation levels and radioactivity in the plant environs. It assists and provides backup support to the detailed effluent monitoring (as recommended in NRC Regulatory Guide 1.21, "Measuring, Evaluating, and Reporting Radioactivity in Solid Wastes and Release of Radioactive Materials in Liquid and Gaseous Effluents from Light-Water Cooled Nuclear Power Plants") which is needed to evaluate individual and population exposures and to verify projected or anticipated radioactivity concentrations.

The applicant plans essentially to continued the proposed preoperational program during the operating period. However, refinements may be made in the program to reflect changes in land use or preoperational monitoring experience.

6.3.5 Meteorological monitoring program

The applicant plans to continue the program begun for the construction permit evaluation. The onsite meteorological tower provides data in accordance with the recommendations of Regulatory Guide 1.23, "Onsite Meteorological Programs." Furthermore, operating technical specifications require meteorological monitoring as a condition of operation.

6.4 RELATED ENVIRONMENTAL RESOURCE DATA

6.4.1 Thermal exception studies

As a condition of the exception to the State Thermal Plan granted by the California Regional Water Quality Control Board, San Diego, the applicants are required to perform studies to determine the optimum mode of heat treatment to control fouling organisms while minimizing adverse effects on marine life and to permit the Regional Board to set precise limits on the frequency, degree, and duration of heat treatment. These studies were submitted to the State Water Resources Control Board on January 31, 1979. On December 18, 1980, the Board determined that the studies fulfilled the conditions set earlier and further determined that the heat treatment operating conditions proposed by the applicant will assure the protection and propagation of a balanced, indigenous population of shellfish, fish, and wildlife within the meaning of Section 316(a) of the Clean Water Act.

6.4.2 Marine Review Committee studies

The California Coastal Commission specified in the Coastal Zone Permit issued in 1974 for SONGS 2 and 3 that an extensive study be conducted at San Onofre. The study program is funded by the utility and is being administered by a three-member Marine Review Committee (MRC) appointed by the Coastal Commission. The intent of the program is to provide an independent assessment of the marine environment and a prediction of the potential impact of SONGS 2 and 3. The MRC has identified the following areas for study: physical, oceanographic, and ecological monitoring and modeling; plankton — far field effects and entrainment; fish populations, impingement, and diversion; and benthic communities, intertidal zone organisms, and kelp beds.

MRC has conducted studies at SONGS 2 and 3 in some of the above mentioned areas since August 1976. In November 1980 the MRC issued a report containing its recommendations, predictions, and rationale. The conclusions of the MRC are essentially consistent with those of the staff as described in Section 5 of this statement. Although noting uncertainties, the MRC has concluded that it does not predict at this time that substantial adverse effects on the marine environment are likely to occur from the operations of the SONGS cooling system. Accordingly, the report recommends no design changes but does recommend continued monitoring of the aquatic community to ensure that there is no serious ecological damage, especially to the kelp beds, as a result of plant operation. (See Appendix E for the options and recommendations of the Marine Review Committee.)

6.5 CONCLUSIONS

The preoperational and operational monitoring programs as described above give adequate attention to impacts discussed in this environmental impact statement.

REFERENCES

These documents are available for inspection and copying for a fee in the NRC Public Document room 1717 H Street, N.W., Washington, D.C. 20555

1. Southern California Edison Company, "San Onofre Nuclear Generating Station Units 2 and 3, Environmental Report - Operating License Stage," Docket No. 50-361/362, 1977.

2. Southern California Edison Company, "San Onofre Generating Station Units 2 and 3, Final Safety Analysis Report," Docket No. 50-361/362, 1977.

3. M. Septoff, A. E. Mitchell, and L. H. Teuscher, "Final Report of the Onshore Tracer Tests Conducted December 1976 through March 1977 at the San Onofre Nuclear Generating Station," Report NUS-1927, NUS Corporation, Rockville, Md., 1977.

7. ENVIRONMENTAL IMPACT OF POSTULATED ACCIDENTS

7.1 PLANT ACCIDENTS

The staff has considered the potential radiological impacts on the environment of possible accidents at the San Onofre Nuclear Generating Station Units 2 and 3 in accordance with a Statement of Interim Policy published by the Nuclear Regulatory Commission on June 13, 1980.[1] The following discussion reflects these considerations and conclusions.

The first section deals with general characteristics of nuclear power plant accidents including a brief summary of safety measures to minimize the probability of their occurrence and to mitigate their consequences if they should occur. Also described are the important properties of radioactive materials and the pathways by which they could be transported to become environmental hazards. Potential adverse health effects and impacts on society associated with actions to avoid such health effects are also identified.

Next, actual experience with nuclear power plant accidents and their observed health effects and other societal impacts are then described. This is followed by a summary review of safety features of the San Onofre Units 2 and 3 facilities and of the site that act to mitigate the consequences of accidents.

The results of calculations of the potential consequences of accidents that have been postulated in the design basis are then given. Also described are the results of calculations for the San Onofre site using probabilistic methods to estimate the possible impacts and the risks associated with severe accident sequences of exceedingly low probability of occurrence.

7.1.1 General characteristics of accidents

The term accident, as used in this section, refers to any unintentional event not addressed in Section 5.5 that results in a release of radioactive materials into the environment. The predominant focus, therefore, is on events that can lead to releases substantially in excess of permissible limits for normal operation. Such limits are specified in the Commission's regulations at 10 CFR Part 20 and 10 CFR Part 50, Appendix I.

There are several features which combine to reduce the risk associated with accidents at nuclear power plants. Safety features in the design, construction, and operation comprising the first line of defense are to a very large extent devoted to the prevention of the release of these radioactive materials from their normal places of confinement within the plant. There are also a number of additional lines of defenses that are designed to mitigate the consequences of failures in the first line. Descriptions of these features for the San Onofre Units 2 and 3 plant may be found in the applicant's Final Safety Analysis Report,[2] and in the staff's Safety Evaluation Report.[3] The most important mitigative features are described in Section 7.1.3.1 below.

These safety features are designed taking into consideration the specific locations of radioactive materials within the plant, their amounts, their nuclear, physical, and chemical properties, and their relative tendency to be transported into and for creating biological hazards in the environment.

7.1.1.1 Fission product characteristics

By far the largest inventory of radioactive material in a nuclear power plant is produced as a byproduct of the fission process and is located in the uranium oxide fuel pellets in the form of fission products. These pellets are contained in the fuel rods which make up the fuel assemblies. During periodic refueling shutdowns, the assemblies containing these fuel pellets are transferred to a spent fuel storage pool so that the second largest inventory of radioactive material is located in this storage area. Much smaller inventories of radioactive materials are also normally present in the water that circulates in the primary coolant system and in the systems used to process gaseous and liquid radioactive wastes in the plant.

These radioactive materials exist in a variety of physical and chemical forms. Their potential for dispersion into the environment is dependent not only on mechanical forces that might physically transport them, but also upon their inherent properties, particularly their volatility. The majority of these materials exist as nonvolatile solids over a wide range of temperatures. Some, however, are relatively volatile solids and a few are gaseous in nature. These characteristics have a significant bearing upon the assessment of the environmental radiological impact of accidents.

The gaseous materials include radioactive forms of the chemcially inert noble gases krypton and xenon. These have the highest potential for release into the atmosphere. If a reactor accident were to occur involving degradation of the fuel cladding, the release of substantial quantities of these radioactive gases from the fuel is a virtual certainty. Such accidents are very low frequency but credible events (see Section 7.1.2). It is for this reason that the safety analysis of each nuclear power plant analyzes a hypothetical design basis accident that postulates the release of the entire contained inventory of radioactive noble gases from the fuel into the containment structure. If further released to the environment as a possible result of failure of safety features, the hazard to individuals from these noble gases would arise predominantly through the external gamma radiation from the airborne plume. The reactor containment structure is designed to minimize this type of release.

Radioactive forms of iodine are formed in substantial quantities in the fuel by the fission process and in some chemical forms may be quite volatile. For this reason, they have traditionally been regarded as having a relatively high potential for release from the fuel. The chemical forms in which the fission product radioiodines are found are generally solid materials at room temperature, however, so that they have a strong tendency to condense (or "plate out") upon cooler surfaces. In addition, most of the iodine compounds are quite soluble in, or chemically reactive with, water. Although these properties do not inhibit the release of radioiodines from degraded fuel, they do act to mitigate the release from containment structures that have large internal surface areas and that contain large quantities of water as a result of an accident. The same properties affect the behavior of radioiodines that may "escape" into the atmosphere. Thus, if rainfall occurs during a release, or if there is moisture on exposed surfaces, e.g., dew, the radioiodines will show a strong tendency to be absorbed by the moisture. Because of radioiodine's relatively high solubility and distinct radiological hazard, its potential for release to the atmosphere has also been reduced by the use of special containment spray systems. If released to the environment, the principal radiological hazard associated with the radioiodines is ingestion into the human body and subsequent concentration in the thyroid gland.

Other radioactive materials formed during the operation of a nuclear power plant have lower volatilities and therefore, by comparison with the noble gases and iodine, a much smaller tendency to escape from degraded fuel unless the temperature of the fuel becomes quite high. By the same token, such materials, if they escape by volatilization from the fuel, tend to condense quite rapidly to solid form again when transported to a lower temperature region and/or dissolve in water when present. The former mechanism can have the result of producing some solid particles of sufficiently small size to be carried some distance by a moving stream of gas or air. If such particulate materials are dispersed into the atmosphere as a result of failure of the containment barrier, they will tend to be carried downwind and deposit on surface features by gravitational settling or by precipitation (fallout), where they will become "contamination" hazards in the environment.

All of these radioactive materials exhibit the property of radioactive decay with characteristic half-lives ranging from fractions of a second to many days or years (see Table 7.1). Many of them decay through a sequence or chain of decay processes and all eventually become stable (nonradioactive) materials. The radiation emitted during these decay processes is the reason that they are hazardous materials.

7.1.1.2 Exposure pathways

The radiation exposure (hazard) to individuals is determined by their proximity to the radioactive material, the duration of exposure, and factors that act to shield the individual from the radiation. Pathways for the transport of radiation and radioactive materials that lead to radiation exposure hazards to humans are generally the same for accidental as for "normal" releases. These are depicted in Section 5, Figure 5.17. There are two additional possible pathways that could be significant for accident releases that are not shown in Figure 5.17. One of these is the fallout onto open bodies of water of radioactivity initially carried in the air. The second would be unique to an accident that results in temperatures inside the reactor core sufficiently high to cause melting and subsequent penetration of the basemat underlying the reactor by the molten core debris. This creates the potential for the release of radioactive material into the hydrosphere through contact with ground water. These pathways may lead to external exposure to radiation, and to internal exposures if radioactivity is inhaled, or ingested from contaminated food or water.

Table 7.1 Activity of Radionuclides in a San Onofre Reactor Core at 3560 MWt

Group/Radionuclide	Radioactive Inventory (millions of curies)	Half-life (days)
A. NOBLE GASES		
Krypton-85	0.63	3,950
Krypton-85m	27	0.183
Krypton-87	52	0.0528
Krypton-88	76	0.117
Xenon-133	190	5.28
Xenon-135	38	0.384
B. IODINES		
Iodine-131	95	8.05
Iodine-132	130	0.0958
Iodine-133	190	0.875
Iodine-134	210	0.0366
Iodine-135	170	0.280
C. ALKALI METALS		
Rubidium-86	0.029	18.7
Cesium-134	8.3	750
Cesium-136	3.3	13.0
Cesium-137	5.2	11,000
D. TELLURIUM-ANTIMONY		
Tellurium-127	0.029	18.7
Tellurium-127m	1.2	109
Tellurium-129	34	0.048
Tellurium-129m	5.9	34.0
Tellurium-131m	14	1.25
Tellurium-132	130	3.25
Antimony-127	6.8	3.88
Antimony-129	37	0.179
E. AKALINE EARTHS		
Strontium-89	100	52.1
Strontium-90	4.1	11,030
Strontium-91	120	0.403
Barium-140	180	12.8
F. COBALT AND NOBLE METALS		
Cobalt-58	0.87	71.0
Cobalt-60	0.32	1,920
Molybdenum-99	180	2.8
Technetium-99m	160	0.25
Ruthenium-103	120	39.5
Ruthenium-105	80	0.185
Ruthenium-106	28	366
Rhodium-105	55	1.50
G. RARE EARTHS, REFRACTORY OXIDES AND TRANSURANICS		
Yttrium-99	4.3	2.67
Yttrium-91	130	59.0
Zirconium-95	170	65.2
Zirconium-97	170	0.71
Niobium-95	170	35.0
Lanthanum-140	180	1.67
Cerium-141	170	32.3
Cerium-143	150	1.38
Cerium-144	95	284

Table 7.1 (Continued)

Group/Radionuclide	Radioactive Inventory (millions of curies)	Half-life (days)
G. <u>RARE EARTHS, REFRACTORY OXIDES AND TRANSURANICS</u> (Continued)		
Praseodymium-143	150	13.7
Neodymium-147	67	11.1
Neptunium-239	1800	2.35
Plutonium-238	0.063	32,500
Plutonium-239	0.023	8.9×10^6
Plutonium-240	0.023	2.4×10^6
Plutonium-241	3.8	5,350
Americium-241	0.0019	1.5×10^5
Curium-242	0.56	163
Curium-244	0.026	6,630

NOTE: The above grouping of radionuclides corresponds to that in Table 7.3.

It is characteristic of these pathways that during the transport of radioactive material by wind or by water, the material tends to spread and disperse, like a plume of smoke from a smokestack, becoming less concentrated in larger volumes of air or water. The result of these natural processes is to lessen the intensity of exposure to individuals downwind or downstream of the point of release, but they also tend to increase the number who may be exposed. For a release into the atmosphere, the degree to which dispersion reduces the concentration in the plume at any downwind point is governed by the turbulence characteristics of the atmosphere which vary considerably with time and from place to place. This fact, taken in conjunction with the variability of wind direction and the presence or absence of precipitation, means that accident consequences are very much dependent upon the weather conditions existing at the time.

7.1.1.3 *Health effects*

The cause and effect relationships between radiation exposure and adverse health effects are quite complex[4] but they have been more exhaustively studied than any other environmental contaminant.

Whole-body radiation exposure resulting in a dose greater than about 10 rem for a few persons and about 25 rem for nearly all people over a short period of time (hours) is necessary before any physiological effects to an individual are clinically detectable. Doses about 10 to 20 times larger than the latter dose, also received over a relatively short period of time (hours to a few days), can be expected to cause some fatal injuries. At the severe, but extremely low probability end of the accident spectrum, exposures of these magnitudes are theoretically possible for persons in the close proximity of such accidents if measures are not or cannot be taken to provide protection, e.g., by sheltering or evacuation.

Lower levels of exposures may also constitute a health risk, but the ability to define a direct cause and effect relationship between any given health effect and a known exposure to radiation is difficult given the backdrop of the many other possible reasons why a particular effect is observed in a specific individual. For this reason, it is necessary to assess such effects on a statistical basis. Such effects include cancer and genetic changes in future generations after exposure of a prospective parent. Cancer in the exposed population may begin to develop only after a lapse of 2 to 15 years (latent period) from the time of exposure and then continue over a period of about 30 years (plateau period). However, in the case of exposure of fetuses (<u>in utero</u>), cancer may begin to develop at birth (no latent period) and end at age 10 (<u>i.e.</u>, the plateau period is 10 years). The health consequences model currently being used is based on the 1972 BEIR Report of the National Academy of Sciences.[5]

Most authorities are in agreement that a reasonable and probably conservative estimate of the statistical relationship between low levels of radiation exposure to a large number of people is within the range of about 10 to 500 potential cancer deaths (although zero

is not excluded by the data) per million man-rem. The range comes from the latest NAS BEIR III Report (1980)[6] which also indicates a probable value of about 150. This value is virtually identical to the value of about 140 used in the current NRC health effects models. In addition, approximately 220 genetic changes per million person-rem would be projected by BEIR III over succeeding generations. That also compares well with the value of about 260 per million man-rem currently used by the NRC staff.

7.1.1.4 Health effects avoidance

Radiation hazards in the environment tend to disappear by the natural process of radioactive decay. Where the decay process is a slow one, however, and where the material becomes relatively fixed in its location as an environmental contaminant (e.g., in soil), the hazard can continue to exist for a relatively long period of time--months, years, or even decades. Thus, a possible consequential environmental societal impact of severe accidents is the avoidance of the health hazard rather than the health hazard itself, by restrictions on the use of the contaminated property or contaminated foodstuffs, milk, and drinking water. The potential economic impacts that this can cause are discussed below.

7.1.2 Accident experience and observed impacts

The evidence of accident frequency and impacts in the past is a useful indicator of future probabilities and impacts. As of mid-1980, there were 69 commercial nuclear power reactor units licensed for operation in the United States at 48 sites with power generating capacities ranging from 50 to 1130 megawatts electric (MWe). (The San Onofre Units 2 and 3 are designed for 1140 MWe each.) The combined experience with these units represents approximately 500 reactor years of operation over an elapsed time of about 20 years. Accidents have occurred at several of these facilities.[7] Some of these have resulted in releases of radioactive material to the environment, ranging from very small fractions of a curie to a few million curies. None is known to have caused any radiation injury or fatality to any member of the public, nor any significant individual or collective public radiation exposure, nor any significant contamination of the environment. This experience base is not large enough to permit a reliable quantitative statistical inference. It does, however, suggest that significant environmental impacts due to accidents are very unlikely to occur over time periods of a few decades.

Melting or severe degradation of reactor fuel has occurred in only one of these 69 operating units, during the accident at Three Mile Island - Unit 2 (TMI-2) on March 28, 1979. In addition to the release of a few million curies of xenon-133, it has been estimated that approximately 15 curies of radioiodine was also released to the environment at TMI-2.[8] This amount represents an extremely minute fraction of the total radioiodine inventory present in the reactor at the time of the accident. No other radioactive fission products were released in measurable quantity.

It has been estimated that the maximum cumulative offsite radiation dose to an individual was less than 100 millirem.[8,9] The total population exposure has been estimated to be in the range from about 1000 to 3000 man-rem. This exposure could produce between none and one additional fatal cancer over the lifetime of the population. The same population receives each year from natural background radiation about 240,000 man-rem and approximately a half-million cancers are expected to develop in this group over its lifetime,[8,9] primarily from causes other than radiation. Trace quantities (barely above the limit of detectability) of radioiodine were found in a few samples of milk produced in the area. No other food or water supplies were impacted.

Accidents at nuclear power plants have also caused occupational injuries and a few fatalities but none attributed to radiation exposure. Individual worker exposures have ranged up to about 4 rems as a direct consequence of accidents, but the collective worker exposure levels (man-rem) are a small fraction of the exposures experienced during normal routine operations that average about 500 man-rem per reactor year.

Accidents have also occurred at other nuclear reactor facilities in the United States and in other countries.[7] Due to inherent differences in design, construction, operation, and purpose of most of these other facilities, their accident record has only indirect relevance to current nuclear power plants. Melting of reactor fuel occurred in at least seven of these accidents, including the one in 1966 at the Enrico Fermi Atomic Power Plant Unit 1. This was a sodium-cooled fast breeder demonstration reactor designed to generate 61 MWe. The damages were repaired and the reactor reached full power four years following the accident. It operated successfully and completed its mission in 1973. This accident did not release any radioactivity to the environment.

A reactor accident in 1957 at Windscale, England released a significant quantity of radioiodine, approximately 20,000 curies, to the environment. This reactor, which was not operated to generate electricity, used air rather than water to cool the uranium fuel. During a special operation to heat the large amount of graphite in this reactor, the fuel overheated and radioiodine and noble gases were released directly to the atmosphere from a 123-m (405-ft) stack. Milk produced in a 512-km^2 (200-mi^2) area around the facility was impounded for up to 44 days. This kind of accident cannot occur in a reactor like San Onofre, however, because of its water-cooled design.

7.1.3 Mitigation of accident consequences

The Nuclear Regulatory Commission is conducting a safety evaluation of the application to operate San Onofre Units 2 and 3. Although this evaluation will contain more detailed information on plant design, the principal design features are presented in the following section.

7.1.3.1 Design features

San Onofre Units 2 and 3 are essentially identical units. Each contains features designed to prevent accidental release of radioactive fission products from the fuel and to lessen the consequences should such a release occur. Many of the design and operating specifications of these features are derived from the analysis of postulated events known as design basis accidents. These accident preventive and mitigative features are collectively referred to as engineered safety features (ESF). The possibilities or probabilities of failure of these systems are incorporated in the assessments discussed in section 7.1.4.

Each steel-lined concrete containment building is a passive mitigating system which is designed to minimize accidental radioactivity releases to the environment. Safety injection systems are incorporated to provide cooling water to the reactor core during an accident to prevent or minimize fuel damage. The containment atmosphere cooling system provides heat removal capability inside the containment following steam release accidents and helps to prevent containment failure due to overpressure. Similarly, the containment spray system is designed to spray cool water into the containment atmosphere. The spray water also contains an additive (sodium hydroxide) which will chemically react with any airborne radioiodine to remove it from the containment atmosphere and prevent its release to the environment.

The mechanical systems mentioned above are supplied with emergency power from onsite diesel generators in the event that normal offsite station power is interrupted.

The fuel handling area of each unit is located in a fuel building, a low leakage structure with a safety-grade ventilation system for accident mitigation. The safety-grade ventilation system is an internal recirculation system and contains both charcoal and high efficiency particulate filters. If radioactivity were to be released into the building, it would be drawn through the ventilation system, and radioactive iodine and particulate fission products would be removed from the flow stream, reducing the concentration within the building and hence the amount that might leak to the atmosphere.

There are features of each unit that are necessary for its power generation function that can also play a role in mitigating certain accident consequences. For example, the main condenser, although not classified as an ESF, can act to mitigate the consequences of accidents involving leakage from the primary to the secondary side of the steam generators (such as steam generator-tube ruptures).

If normal offsite power is maintained, the ability of the plant to send contaminated steam to the condenser instead of releasing it through the safety valves or atmospheric dump valves can significantly reduce the amount of radioactivity released to the environment. In this case, the fission product removal capability of the normally operating off-gas treatment system would come into play.

Much more extensive discussions of the safety features and characteristics of San Onofre Units 2 and 3 may be found in the applicant's Final Safety Analysis Report.[2] The staff evaluation of these features is addressed in the Safety Evaluation Report. In addition, the implementation of the lessons learned from the TMI-2 accident, in the form of improvements in design and procedures, and operator training, will significantly reduce the likelihood of a degraded core accident which could result in large releases of fission products to the containment. Specifically, the applicant will be required to meet those TMI-related requirements specified in NUREG-0737. As noted in Section 7.1.4.7, no credit has been taken for these actions and improvements in discussing the radiological risk of accidents in this supplement.

7.1.3.2 Site features

In the process of considering the suitability of the site of San Onofre Units 2 and 3, pursuant to NRC's Reactor Site Criteria in 10 CFR Part 100, consideration was given to certain factors that tend to minimize the risk and the potential impact of accidents. First, the site has an exclusion area as required in 10 CFR Part 100. The exclusion area of the (33.8 hectare (83.6-acre)) site has a minimum exclusion distance of (1968 ft) 600 meters from the containment centerlines to the closest site boundary. The applicant's authority to control all activities within the exclusion area was acquired by a grant of easement from the United States of America made by the Secretary of the Navy. The exclusion area is traversed by old U.S. Highway 101, the San Diego Freeway (Interstate 5), and the Atchison, Topeka and Santa Fe Railroad. The exclusion area on the ocean side extends over a narrow strip of beach and into the Pacific Ocean.

The applicant's control of the landward portion of the exclusion area extends to the mean high tide line but does not include the strip of beach lying between high and low tide that is occasionally uncovered. This strip of "tidal beach" is owned by the State of California and is used primarily as a passageway for individuals walking along the beach. The applicant's lack of control of this strip of tidal beach has been adjudicated in a Commission proceeding (see ALAB-432) and has been determined to be de minimis on the basis of its occasional use, together with the high probability that any radiation exposure to individuals in this zone will be within the guideline values of 10 CFR Part 100 in the event of an emergency.

Activities within the exclusion area which are unrelated to plant operation include a gas pipeline, railroad traffic, through traffic on the San Diego Freeway, and local recreational traffic on old U.S. Highway 101. Recreational activities in the plant vicinity include swimming, camping, and surfing. Recreational activities, such as sunbathing or picnicking, are discouraged within the landward portion of the exclusion area (the area landward of the contour of mean high tide). The seaward portion of the exclusion area (the area seaward of the contour of mean high tide) may be occupied by small numbers of people for passageway transit between the public beach areas upcoast and downcoast from the plant. Additional small numbers of people may be anticipated to occasionally be in the water within the exclusion area.

Transient access to an approximate 2.02-hectare (5-acre) at the southwest corner of the site for the purposes of viewing the scenic bluffs and barrancas will be on an unimproved walkway. The applicant has estimated that at any one time a maximum of 100 persons will be in the walkway and a 2.02-hectare (5-acre) viewing area, and on the beach and water below the mean high tide. The improved walkway affords landward passage between the two beach areas.

In case of a radiological emergency, the applicants have made arrangements with agencies of the State and local governments to control all traffic on the railroad, roadways, and waterways.

Second, beyond and surrounding the exclusion area is a low population zone (LPZ), also required by Part 100. This is a circular area of 3.14 km (1.95 mi) outer radius. Within this zone the applicant must assure that there is reasonable probability that appropriate measures could be taken on behalf of the residents in the event of a serious accident.

The San Onofre State Beach northwest and southwest of the San Onofre exclusion areas represents a public waterfront recreation area within an 8-km (5-mi) radius of the plant. The beach south of the nuclear facility is used for swimming, hiking, and vehicle parking. The 1036 m (3,400-ft) stretch of beach north of the site is used primarily for surfing.

The largest communities in the vicinity of the site are San Clemente, located about 4.8 km (3 mi) away, which had a 1976 estimated population of 23,000, and the U.S. Marine Corp base Camp Pendleton, with a total estimated population of about 33,000. The Marine Corp base consists of several population clusters or camps located at distances from 2.4 km to 19.31 km (1.5 mi to 12 mi) away.

The applicant has estimated a peak transient population in major tourist and recreational activities along Interstate 5 in a 16-km (10-mi) radius of the plant to be 56,600 persons. This occurs during the summer months and is due to persons engaged in water sport recreation on the Pacific Ocean beach and coastal waters.

The Mexican border lies about 121 km (75 mi) from San Onofre, toward the southeast. The cities of Tijuana, Mexicali, and Ensenada are within 241 km (150 mi) of the site.

The safety evaluation of the San Onofre site has also included a review of potential external hazards, i.e., activities offsite that might adversely affect the operation of the plant and cause an accident. This review encompassed nearby industrial, transportation, and military facilities that might create explosive, missile, toxic gas, or similar hazards. The staff concluded at the construction permit stage that the hazards from the nearby military facility are negligibly small. However, the hazards from the nearby interstate highway, the railroad right of way, and natural gas pipelines, are still under review by the staff. Reevaluation of these hazards has been requested by the staff, and the results will be reported in a supplement to the staff's Safety Evaluation Report. It is anticipated that the review will show that either the risks are acceptably small or may be acceptably small.

7.1.3.3 Emergency preparedness

Emergency preparedness plans including protective action measures for the San Onofre facility and environs are in an advanced, but not yet fully completed stage. In accordance with the provisions of 10 CFR Section 50.47, effective November 3, 1980, no operating license will be issued to the applicant unless a finding is made by the NRC that the state of onsite and offsite emergency preparedness provides reasonable assurance that adequate protective measures can and will be taken in the event of a radiological emergency. Among the standards that must be met by these plans are provisions for two Emergency Planning Zones (EPZ). A plume exposure pathway EPZ of about 16 km (10 mi) in radius and an ingestion exposure pathway EPZ of about 80 km (50 mi) in radius are required. Other standards include appropriate ranges of protective actions for each of these zones, provisions for dissemination to the public of basic emergency planning information, provisions for rapid notification of the public during a serious reactor emergency, and methods, systems, and equipment for assessing and monitoring actual or potential offsite consequences in the EPZs of a radiological emergency condition.

NRC findings will be based upon a review of the Federal Emergency Management Agency (FEMA) findings and determinations as to whether State and local government emergency plans are adequate and capable of being implemented, and on the NRC assessment as to whether the applicant's onsite plans are adequate and capable of being implemented. NRC staff findings will be reported in a supplement to the staff's Safety Evaluation Report. Although the presence of adequate and tested emergency plans cannot prevent the occurrence of an accident, it is the judgment of the staff that they can and will substantially mitigate the consequences to the public if one should occur.

7.1.4 Accident risk and impact assessment

7.1.4.1 Design basis accidents

As a means of assuring that certain features of the San Onofre Units 2 and 3 plants meet acceptable design and performance criteria, both the applicant and the staff have analyzed the potential consequences of a number of postulated accidents. Some of these could lead to significant releases of radioactive materials to the environment, and calculations have been performed to estimate the potential radiological consequences to persons offsite. For each postulated initiating event, the potential radiological consequences cover a considerable range of values depending upon the particular course taken by the accident and the conditions, including wind direction and weather, prevalent during the accident.

In the safety analysis of the San Onofre Units 2 and 3 plants, three categories of accidents have been considered. These categories are based upon their probability of occurrence and include (a) incidents of moderate frequency, i.e., events that can reasonably be expected to occur during any year of operation, (b) infrequent accidents, i.e., events that might occur once during the lifetime of the plant, and (c) limiting faults, i.e., accidents not expected to occur but that have the potential for significant releases of radioactivity. The radiological consequences of incidents in the first category, also called anticipated operational occurrences, are discussed in Section 5. Initiating events postulated in the second and third categories for the San Onofre Units 2 and 3 are shown in Table 7.2. These are collectively designated design basis accidents in that specific design and operating features as described above in Section 7.1.3.1 are provided to limit their potential radiological consequences. Approximate radiation doses that might be received by a person at the nearest site boundary (600 meters from the plant) are also shown in the table, along with a characterization of the time duration of the releases.

Table 7.2 Approximate Radiation Doses from Design Basis Accidents, Conservative Calculational Model

	Duration of Release	Dose (rem at 600 m[1]) Whole Body	Thyroid
Infrequent Accidents:			
Waste Gas Tank Failure	< 2 hr	< 3	< 30
Steam Generator Tube[2] Rupture	< 2 hr	< 3	2
Fuel Handling Accident	< 2 hr	7	40
Limiting Faults:			
Main Steam Line Break	< 2 hr	6	10
Control Rod Ejection	hrs-days	< 6	60
Large-Break LOCA	hrs-days	3	100

[1]The nearest site boundary.
[2]See NUREG-0651[6] for descriptions of three steam generator tube rupture accidents that have occurred in the United States.

The calculational model used is a conservative one in that it is expected to provide a reasonable estimate of the potential upper bound for individual exposures. The results are used to implement the provisions of 10 CFR 100 and to establish performance requirements for certain engineered safety features. The conservative assumptions used in these analyses include: (1) large (upper bound) amounts of radioactive material released by the initiating event, (2) single failures in important equipment, including operating the engineered safety features in a degraded mode,* (3) very adverse meteorological conditions, and (4) no reduction in exposure due to possible protective actions.

The results of these calculations show that, for these events, the limiting whole-body exposures are not expected to exceed 7 rem. They also show that radioiodine releases have the potential for offsite exposures ranging up to about 100 rem to the thyroid. For such an exposure to occur, an individual would have to be located at a point on the site boundary where the radioiodine concentration in the plume has its highest value and inhale at a breathing rate characteristic of a person jogging, for a period of two hours. The health risk to an individual receiving such a thyroid exposure is the potential appearance of benign or malignant thyroid nodules in about 4 out of 100 cases, and the development of a fatal cancer in about 2 out of 1000 cases.

The realistically expected consequences, were one of these initiating events actually to occur, would be very substantially less. Therefore, the risk is judged to be extremely small for these design basis accidents. The subject of risk is more fully discussed in Section 7.1.4.6 below.

7.1.4.2 Probabilistic assessment of severe accidents

In this and the following three sections, there is a discussion of the probabilities and consequences of accidents of greater severity than the design basis accidents identified in the previous section. As a class, they are considered less likely to occur, but their consequences could be more severe, both for the plant itself and for the environment. These severe accidents, heretofore frequently called Class 9 accidents, can be distinguished from design basis accidents in two primary respects: they involve substantial physical deterioration of the fuel in the reactor core, including overheating to the point of melting, and they involve deterioration of the capability of the containment structure to perform its intended function of limiting the release of radioactive materials to the environment.

*The containment structure, however, is assumed to prevent leakage in excess of that which can be demonstrated by testing, as provided in 10 CFR Section 100.11(a).

The assessment methodology employed is that described in the Reactor Safety Study (RSS) which was published in 1975.[10*] The San Onofre Units 2 and 3 are Combustion Engineering-designed pressurized water reactors (PWR) having similar design and operating characteristics to the Surry Unit 1 facility used in the RSS as a prototype for PWRs. This assessment has used as its starting point, therefore, the same set of accident sequences that were found in the RSS to be dominant contributors to risk in the prototype PWR. The same set of nine release categories, designated PWR 1 through 9, have also been used to represent the spectrum of severe accident releases that are hypothesized for the San Onofre Units 2 and 3. Characteristics of these categories are shown in Table 7.3. Sequences initiated by natural phenomena such as tornadoes, floods, or seismic events and those that could be initiated by deliberate acts of sabotage are not included in these events sequences. The radiological consequences of such events would not be different in kind from those which have been treated. Moreover, it is the staff's judgment, based upon design requirements of 10 CFR Part 50, Appendix A, relating to effects of natural phenomena, and safeguards requirements of 10 CFR Part 73, that these events do not contribute significantly to risk. The facts upon which the staff based its Safe Shutdown Earthquake and its conclusions regarding the effects of natural phenomena on the plant are given in the Safety Evaluation Report.

A calculated probability per reactor year associated with each release category is also shown in the second column in Table 7.3. These probabilities are the result of a detailed engineering analysis of the prototype PWR in the Reactor Safety Study. There are substantial uncertainties in these probabilities. This is due, in part, to difficulties associated with the quantification of human error and to inadequacies in the data base on failure rates of individual plant components that were used to calculate the probabilities[11] (see Section 7.1.4.7 below). Also, the detailed engineering analysis represents a plant designed by a different nuclear steam supply system designer (CE versus Westinghouse) with different detailed designs. The probability of accident sequences from the Surry plant were used to give a perspective of the societal risk at San Onofre Units 2 and 3 because, although the probabilities of particular accident sequences may be substantially different, the overall effect of all sequences taken together is likely to be within the uncertainties. Except as indicated in the footnotes in Table 7.3, the staff has no present basis for judging whether the probabilities may be too high or too low. The error band for the probabilities of some of the event sequences could be as great as a factor of 100. The event sequences in categories PWR 1-7 lead to partial or complete melting of the reactor core while those in the last two categories do not involve melting of the core. In release categories 1 to 3, the event sequences include containment failure by steam explosion, hydrogen burning, or overpressure. In release categories 6 and 7, the dominant containment failure mode is by melt-through of the containment base mat. The other release categories contain event sequences in which the systems intended to isolate the containment fail to act properly.

The magnitudes (curies) of radioactivity releases for each category are obtained by multiplying the release fractions shown in Table 7.3 by the amounts that would be present in the core at the time of the hypothetical accident. These are shown in Table 7.1 for a San Onofre plant at a core thermal power level of 3560 megawatts.

The potential radiological consequences of these releases have been calculated by the consequence model used in the RSS[12] and adapted to apply to a specific site. The essential elements are shown in schematic form in Figure 7.1. Environmental parameters specific to the San Onofre site have been used and include the following:

(1) Meteorological data for the site representing a full year of consecutive hourly measurements and seasonal variations.

(2) Projected population in the United States and Mexico for the year 2000 extending throughout regions of 80 and 560 km (50 and 350 mi) radius from the site.

(3) The habitable land fraction within the 560-km (350-mi) radius.

(4) Land use statistics, on a state-wide basis, including farm land values, farm product values including dairy production, and growing season information, for the State of California and each surrounding State within the 560-km (350-mi) region.

(5) Land use statistics for Mexico on a country-wide basis. Farm land values, growing season information, and comparison between agriculture and dairy products are based on comparison with U.S. values for nearby States. Farm product values are based on Mexico-average Gross National Product and "agriculture" percentage.

*Because this report has been the subject of considerable controversy, a discussion of the uncertainties surrounding it is provided in Section 7.1.4-7.

Table 7.3

Summary of Atmospheric Release Categories Representing Hypothetical Accidents in a PWR

Release Category	Probability (reactor-yr^{-1})	Fraction of Core Inventory Released[a]						
		Xe-Kr	I	Cs-Rb	Te-Sb	Ba-Sr	Ru[b]	La[c]
PWR 1	5.1×10^{-8}[d]	0.9	0.7	0.4	0.4	0.05	0.4	3×10^{-3}
PWR 2	7×10^{-6}	0.9	0.7	0.5	0.3	0.06	0.02	4×10^{-3}
PWR 3	2.3×10^{-6}	0.8	0.2	0.2	0.3	0.02	0.03	3×10^{-3}
PWR 4	2.1×10^{-11}	0.6	0.09	0.04	0.03	5×10^{-3}	3×10^{-3}	4×10^{-4}
PWR 5	5×10^{-8}	0.3	0.03	9×10^{-3}	5×10^{-3}	1×10^{-3}	6×10^{-4}	7×10^{-5}
PWR 6	6×10^{-7}	0.3	3×10^{-3}	8×10^{-4}	1×10^{-3}	9×10^{-5}	7×10^{-5}	1×10^{-5}
PWR 7	4×10^{-5}	6×10^{-3}	4×10^{-5}	1×10^{-5}	2×10^{-5}	1×10^{-6}	1×10^{-6}	2×10^{-7}
PWR 8	4×10^{-5}	2×10^{-3}	1×10^{-4}	5×10^{-4}	1×10^{-6}	1×10^{-8}	0	0
PWR 9	4×10^{-4}	3×10^{-6}	1×10^{-7}	6×10^{-7}	1×10^{-9}	1×10^{-11}	0	0

(a) Background on the isotope groups and release mechanisms is presented in Appendix VII, WASH-1400 (Ref. 9).

(b) Includes Ru, Rh, Co, Mo, Tc.

(c) Includes, Y, La, Zr, Nb, Ce, Pr, Nd, Np, Pu, Am, Cm.

(d) Current understanding of the phenomenon of containment failure by steam explosion embodied in this release category indicates the probability should be lower than stated.

NOTE: Refer to Section 7.1.4.6 for a discussion of uncertainties in risk estimates.

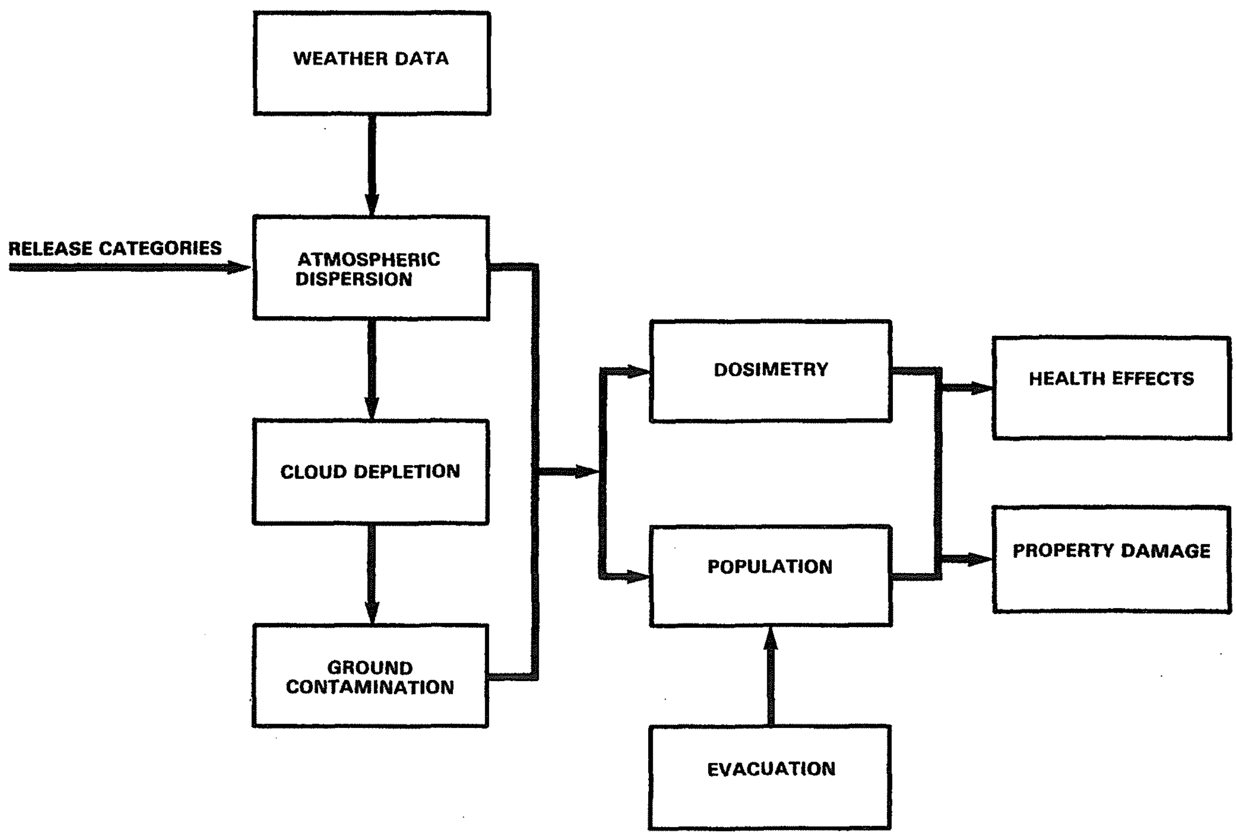


Figure 7.1 Schematic outline of consequence model

To obtain a probability distribution of consequences the calculations are performed assuming the occurrence of each accident release sequence at each of 91 different "start" times throughout a one-year period. Each calculation utilizes the site specific hourly meteorological data and seasonal information for the time period following each "start" time. The consequence model also contains provisions for incorporating the consequence reduction benefits of evacuation and other protective actions. Early evacuation of people would considerably reduce the exposure from the radioactive cloud and the contaminated ground in the wake of the cloud passage. The evacuation model used (see Appendix F) has been revised from that used in the RSS for better site-specific application. The quantitative characteristics of the evacuation model used for the San Onofre site are best estimate values made by the staff and based upon evacuation time estimates prepared by the applicant. Actual evacuation effectiveness could be greater or less than that characterized but would not be expected to be much less, even under adverse conditions.

The other protective actions include: (a) either complete denial of use (interdiction), or permitting use only at a sufficiently later time after appropriate decontamination of food stuffs such as crops and milk, (b) decontamination of severely contaminated environment (land and property) when it is considered to be economically feasible to lower the levels of contamination to protective action guide (PAG) levels, and (c) denial of use (interdiction) of severely contaminated land and property for varying periods of time until the contamination levels reduce to such values by radioactive decay and weathering so that land and property can be economically decontaminated as in (b) above. These actions would reduce the radiological exposure to the people from immediate and/or subsequent use of or living in the contaminated environment.

Early evaucation and other protective actions as mentioned above are considered as essential sequels to serious nuclear reactor accidents involving significant release of radioactivity to the atmosphere. Therefore, the results shown for San Onofre reactor include the benefits of these protective actions.

There are also uncertainties in the estimates of consequences, and the error bounds may be as large as they are for the probabilities. It is the judgment of the staff, however, that it is more likely that the calculated results are overestimates of consequences rather than underestimates.

The results of the calculations using this consequence model are radiological doses to individuals and to populations, health effects that might result from these exposures, costs of implementing protective actions, and costs associated with property damage by radioactive contamination.

7.1.4.3 Dose and health impacts of atmospheric releases

The results of the calculations of dose and health impacts performed for the San Onofre facility and site are presented in the form of probability distributions in Figures 7.2 through 7.5 and are included in the impact Summary Table 7.4. All of the nine release categories shown in Table 7.3 contribute to the results, the consequences from each being weighted by its associated probability.

Figure 7.2 shows the probability distribution for the number of persons who might receive whole body doses equal to or greater than 200 rem and 25 rem, and thyroid doses equal to or greater than 300 rem from early exposure,* all on a per-reactor-year basis. The 200 rem whole body dose figure corresponds approximately to a threshold value for which hospitalization would be indicated for the treatment of radiation injury. The 25 rem whole body (which has been identified earlier as the lower limit for clinically obervable physiological effects in nearly all people) and 300 rem thyroid figures correspond to the Commission's guidelines values for reactor siting in 10 CFR Part 100.

The figure shows in the left-hand portion that there is less than 1 chance in 100,000 per year (i.e. 10^{-5}) that one or more persons may receive doses equal to or greater than any of the doses specified. The fact that the three curves run almost parallel in horizontal lines initially shows that if one person were to receive such doses, the chances are about the same that several tens to hundreds would be so exposed. The chances of larger numbers of persons being exposed at those levels are seen to be considerably smaller. For example, the chances are about 1 in 100,000,000 (i.e. 10^{-8}) that 100,000 or more people might receive doses of 200 rem or greater. A majority of the exposures reflected in this figure would be expected to occur to persons within a 80-km (50-mi) radius of the plant. Virtually all would occur with a 160-km (100-mi) radius.

*The containment structure, however, is assumed to prevent leakage in excess of that which can be demonstrated by testing, as provided in 10 CFR Section 100.11(a).

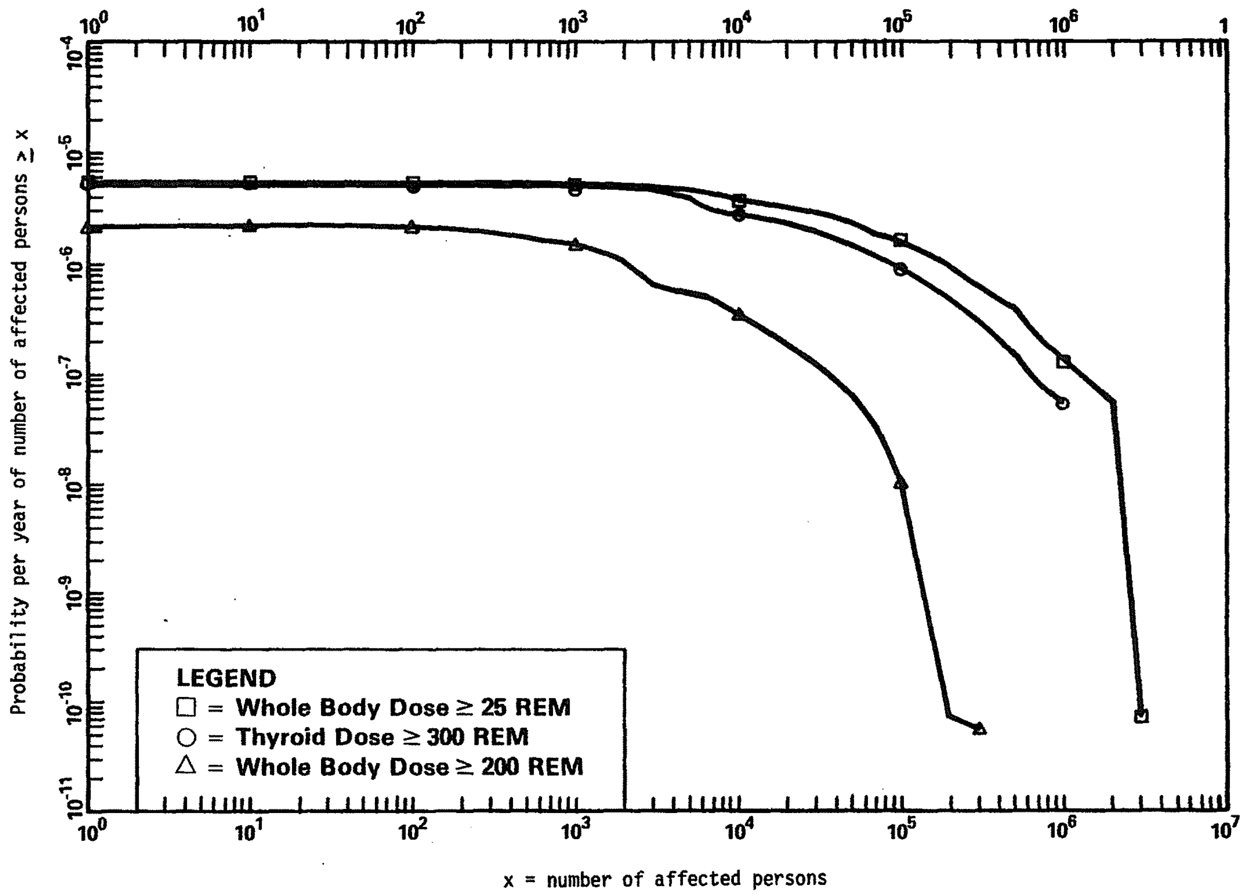


Figure 7.2 Probability distribution of individual dose impacts
(See Section 7.1.4.6 for a discussion of uncertainties in risk estimates.)

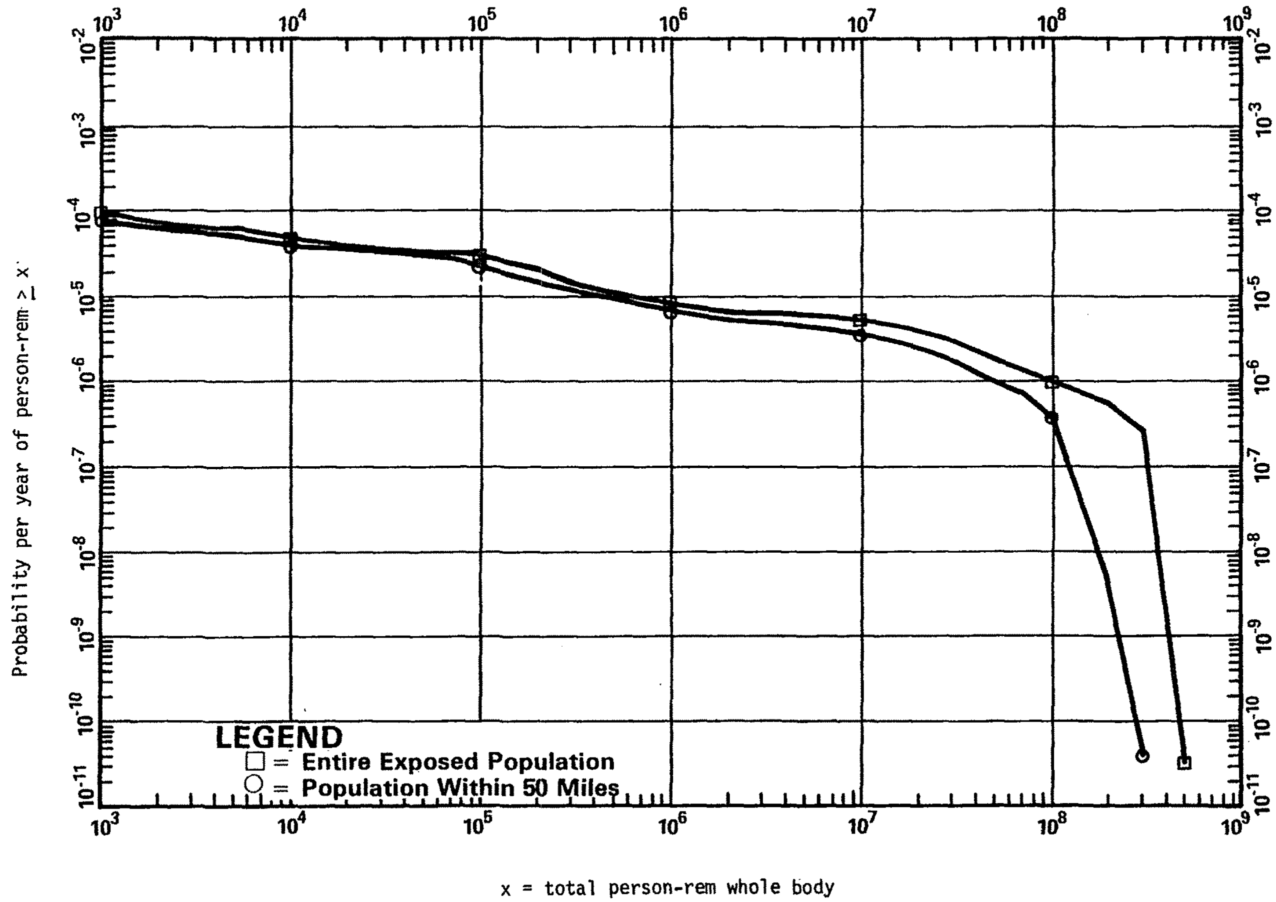


Figure 7.3 Probability distribution of population exposures. (See Section 7.1.4.6 for discussion of uncertainties in risk estimates.) (To convert miles to kilometers, multiply by 1.6.)

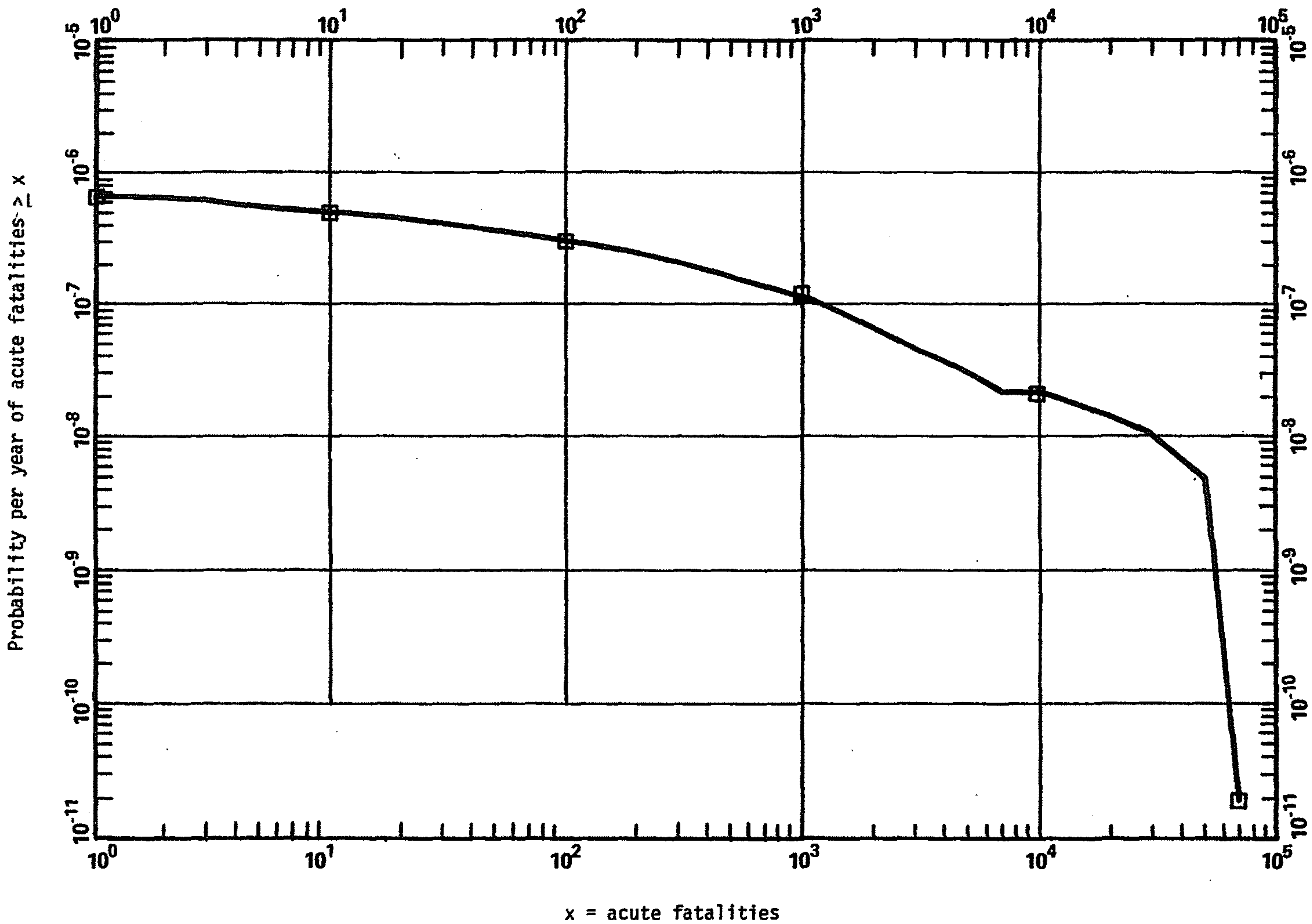


Figure 7.4 Probability distribution of acute fatalities. (See Section 7.1.4.6 for discussion of uncertainties in risk estimates.)

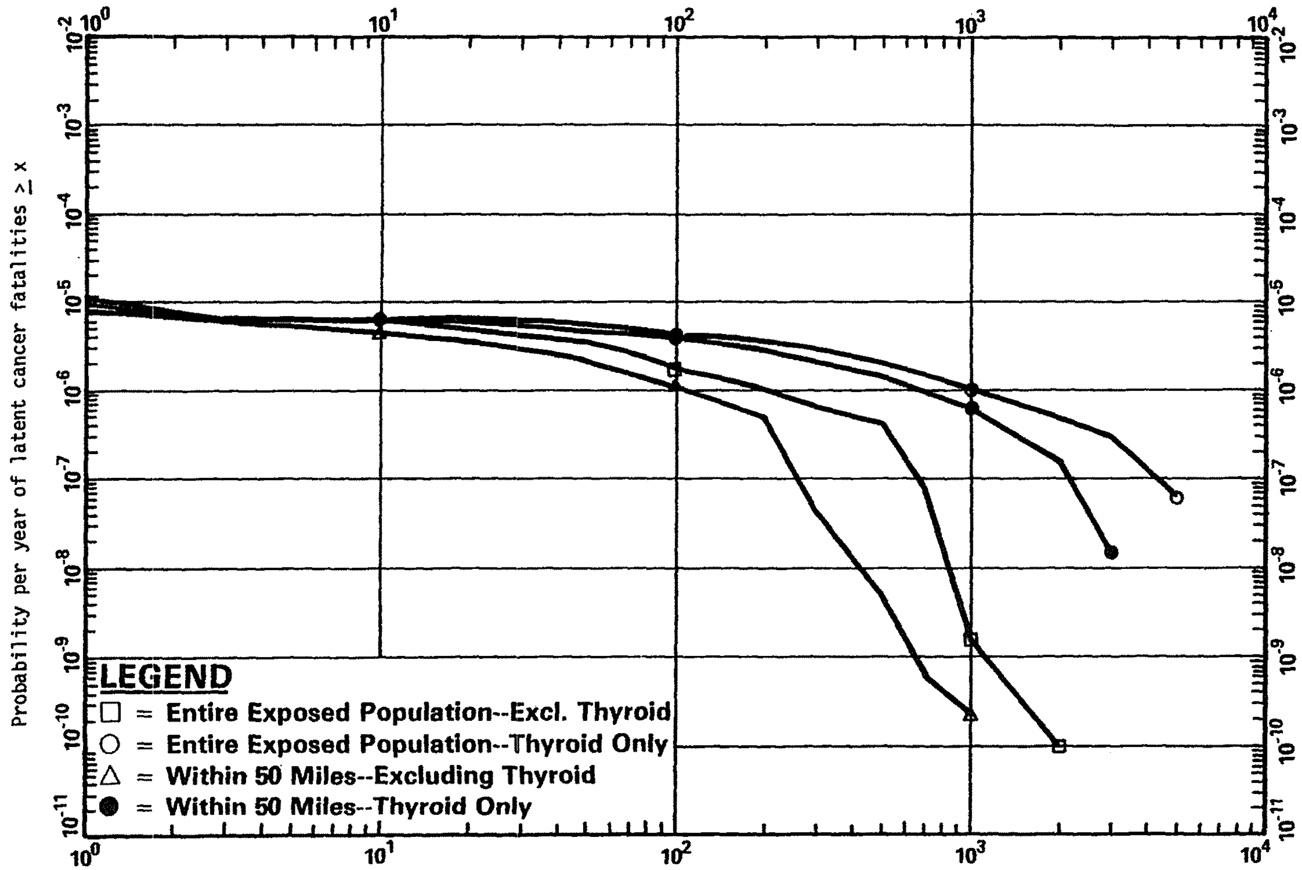


Figure 7.5 Probability distribution of cancer fatalities. (See Section 7.1.4.6 for discussion of uncertainties in risk estimates.) (To convert miles to kilometers, multiply by 1.6.)

Table 7.4 Summary of Environmental Impacts and Probabilities

Probability of impact per year	Persons exposed over 200 rem	Persons exposed over 25 rem	Acute fatalities	Population exposure, millions of man-rem 80 km/total	Latent* cancers, 80 km/ total	Cost of offsite mitigating actions, $ millions
10^{-4}	< 1	< 1	< 1	< 0.001	< 60	< 0.001
10^{-5}	< 1	< 1	< 1	0.4/0.6	< 60	12
5×10^{-6}	< 1	160	< 1	2/10	1,400/2,500	400
10^{-6}	2,000	190,000	< 1	45/100	23,000/36,000	5,000
10^{-7}	31,000	1,100,000	1,100	110/300	71,000/143,000	15,000
10^{-8}	100,000	2,000,000	30,000	170/340	12,000/24,000	35,000
Related Figure	7.2	7.2	7.4	7.3	7.5	7.6

*Genetic effects would be approximately twice the number of latent cancers. Thirty times the values shown in the Figure 7.5 are shown in this column reflecting the 30-year period over which they might occur.

NOTE: Refer to Section 7.1.4.6 for a discussion of uncertainties in risk estimates.

Figure 7.3 shows the probability distribution for the total population exposure in person-rem, i.e., the probability per reactor-year that the total population exposure will equal or exceed the values given. A substantial fraction of the population exposure would occur within 80 km (50 mi) but the more severe releases (PWR 1-6) would result in exposure to persons beyond the 80-km (50-mi) range as shown.

For perspective, population doses shown in Figure 7.3 may be compared with the annual average dose to the population within 80 km (50 mi) of the San Onofre site due to natural background radiation of 700,000 man-rem, and to the anticipated annual population dose to the general public from normal station operation of 460 man-rem (excluding plant workers) (Section 5, Table 5.3 and 5.5).

Figure 7.4 shows the probability distribution for acute fatalities, representing radiation injuries that would produce fatalities within about one year after exposure. Virtually all of the acute fatalities would be expected to occur within a 64-km (40-mi) radius. The results of the calculations shown in this figure and in Table 7.4 reflect the effect of evacuation within the 16-km (10-mi) plume exposure pathway EPZ only. For the very low probability accidents having the potential for causing radiation exposure above the threshold for acute fatality at distances beyond 16 km (10 mi), it would be realistic to expect that authorities would evacuate persons at all distances at which such exposures might occur. Acute fatality consequences would therefore reasonably be expected to be very much less than the numbers shown.

Figure 7.5 represents the statistical relationship between population exposure and the induction of fatal cancers that might appear over a period of many years following exposure. The impacts on the total population and the population within 80 km (50 mi) are shown separately. Further, the fatal, latent cancers have been subdivided into those attributable to exposures of the thyroid and all other organs.

7.1.4.4 Economic and societal impacts

As noted in Section 7.1.1, the various measures for avoidance of adverse health effects including those due to residual radioactive contamination in the environment are possible consequential impacts of severe accidents. Calculations of the probabilities and magnitudes of such impacts for the San Onofre facility and environs have also been made. Unlike the radiation exposure and adverse health effect impacts discussed above, impacts associated with adverse health effects avoidance are more readily transformed into economic impacts.

The results are shown as the probability distribution for costs of offsite mitigating actions in Figure 7.6 and are included in the impact Summary Table 7.4. The factors contributing to these estimated costs include the following:

- Evacuation costs
- Value of crops contaminated and condemned
- Value of milk contaminated and condemned
- Costs of decontamination of property where practical
- Indirect costs due to loss of use of property and incomes derived therefrom.

The last named costs would derive from the necessity for interdiction to prevent the use of property until it is either free of contamination or can be economically decontaminated.

Figure 7.6 shows that at the extreme end of the accident spectrum these costs could exceed tens of billions of dollars but that the probability that this would occur is exceedingly small, less than one chance in a hundred million per year.

Additional economic impacts that can be monetized include costs of decontamination of the facility itself and the costs of replacement power. Probability distributions for these impacts have not been calculated, but they are included in the discussion of risk considerations in Section 7.1.4.6 below.

7.1.4.5 Releases to groundwater

A pathway for public radiation exposure and environmental contamination that could be associated with severe reactor accidents was identified in Section 7.1.1.2 above. Consideration has been given to the potential environmental impact of this pathway for the San Onofre plant. The principal contributors to the risk are the core melt accidents associated with the PWR-1 through 7 release categories. The penetration of the basemat of the

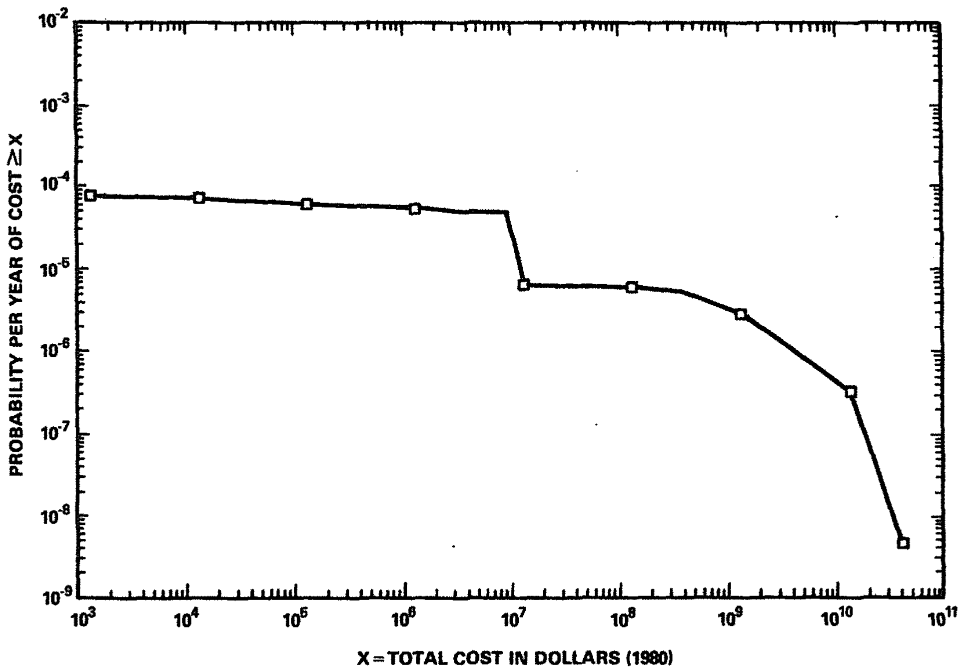


Figure 7.6 Probability distribution of cost of offsite mitigative measures

containment building can release molten core debris to the strata beneath the plant. Soluble radionuclides in this debris can be leached and transported with groundwater to downgradient domestic wells used for drinking or to surface water bodies used for drinking water, aquatic food and recreation. In pressurized water reactors, such as the San Onofre unit, there is an additional opportunity for groundwater contamination due to the release of contaminated sump water to the ground through a breach in the containment.

An analysis of the potential consequences of a liquid pathway release of radioactivity for generic sites was presented in the "Liquid Pathway Generic Study" (LPGS).[13] The LPGS compared the risk of accidents involving the liquid pathway (drinking water, irrigation, aquatic food, swimming and shoreline usage) for four conventional, generic land-based nuclear plants and a floating nuclear plant, for which the nuclear reactors would be mounted on a barge and moored in a water body. Parameters for the land-based sites were chosen to represent averages for a wide range of real sites and are thus "typical," but represented no real site in particular.

The discussion in this section is an analysis to determine whether or not the San Onofre site liquid pathway consequences would be unique when compared to land-based sites considered in the LPGS. The method consists of a direct scaling of the LPGS population doses based on the relative values of key parameters characterizing the LPGS "ocean" site and the San Onofre site. The parameters which were evaluated included amounts of radioactive materials entering the ground, groundwater travel time, sorption on geological media, surface water transport, aquatic food consumption, and shoreline usage.

Doses to individuals and populations were calculated in the LPGS without consideration of interdiction methods such as isolating the contaminated groundwater or denying use of the water. In the event of surface water contamination, commercial and sports fishing, as well as many other water-related activities would be restricted. The consequences would therefore be largely economic or social, rather than radiological. In any event, the individual and population doses for the liquid pathway range from fractions to very small fractions of those that can arise from the airborne pathways.

The San Onofre reactors are situated above the San Mateo Formation, which is about 274-m (876.8-ft) thick and consists of medium to coarse-grained sandstone.[2] Groundwater at the site occurs between elevation 0 and 1.5 m (4.8 ft) Mean Low-Low Water, under water table conditions. The basemat of the reactors would be beneath the water table.

The groundwater gradient is clearly toward the ocean. There are no wells between the site and the ocean, so no groundwater users could be affected by an accidental contamination from the plant. There is virtually no possibility of a reversal of the groundwater gradient due to heavy pumping inland, particularly because such a reversal would at the same time cause an unacceptable intrusion of saltwater into the aquifer. Therefore, liquid radioactivity released from a core melt accident could only cause contamination by being transported through the groundwater and subsequently released to the Pacific Ocean.

The staff's most conservative estimate of the groundwater travel time would be 215 days. For groundwater travel times of this magnitude, it is clear that the most important radionuclide contributors to the liquid pathway population dose would be Sr-90 and Cs-137. Conservative values of the retardation factors, which reflect the effects of sorption of the radionuclides on geologic materials, were estimated on media similar to the granular materials under the site[14] to be 31 for Sr-90 and 2204 for Cs-137. The mean transport time from the reactor building to the Pacific Ocean is therefore conservatively estimated to be about 16 years for Sr-90 and 1080 years for Cs-137. When these travel times are compared to 5.7 years for Sr-90 and 51 years for Cs-137 in the LPGS land-based ocean site case, the relatively larger travel times for the San Onofre site would allow a smaller portion of the radioactivity to enter the surface water. This reduces the Sr-90 release to about 78% of the LPGS value. Virtually all of the Cs-137 would have decayed before reaching surface water.

Contaminants released from the shoreline would disperse in the oceanic turbulence. The LPGS made no distinction between the turbulence which would be found in the east, gulf, or west coasts of the United States. The only assumption which can be made without site-specific data is that the mixing at the San Onofre and LPGS sites are similar.

The two major liquid pathway exposure pathways for an ocean site are aquatic food consumption and direct shoreline exposure.

The commercial and recreational finfish harvest for a rectangular block 80 km along shore and stretching 40 km offshore has been estimated by the staff from data provided in the

Environmental Report[15] to be about 13.1×10^6 kg. For comparison, the same size block using the LPGS ocean site fish catch densities would yield 5.8×10^6 kg of finfish.

Approximately 62% of population dose due to finfish consumption calculated in the LPGS was due to Cs-137 and approximately 38% was due to Sr-90. The only significant radionuclide which could reach the ocean in the San Onofre case would be Sr-90. The staff has conservatively estimated that the uninterdicted population dose in the San Onofre case would be about 69% of the LPGS land-based ocean case population dose for seafood consumption.

Nearly all of the direct shoreline exposure in the LPGS ocean-based site case was determined to emanate from Cs-137. Since virtually all of the Cs-137 would decay before reaching the ocean, the shoreline direct exposure can be eliminated from further consideration.

The San Onofre liquid pathway contribution to population dose has, therefore, been demonstrated to be smaller than that predicted for the LPGS land-based ocean site, which represents a "typical" ocean site. Thus, the San Onofre site is not unique in its liquid pathway contribution to risk.

There are measures which could be taken to minimize the impact of the liquid pathway. The staff estimated that the minimum groundwater travel time from the San Onofre site to the Pacific Ocean would be hundreds of days. In addition, the holdup of important radionuclides would provide additional time to utilize engineering measures such as slurry walls and well-point dewatering to isolate the radioactive contaminants at the source.

7.1.4.6 Risk considerations

The foregoing discussions have dealt with both the frequency (or likelihood of occurrence) of accidents and their impacts (or consequences). Since the ranges of both factors are quite broad, it is useful to combine them to obtain average measures of environmental risk. Such averages can be particularly instructive as an aid to the comparison of radiological risks associated with accident releases and with normal operational releases.

A common way in which this combination of factors is used to estimate risk is to multiply the probabilities by the consequences. The resultant risk is then expressed as a number of consequences expected per unit of time. Such a quantification of risk does not at all mean that there is universal agreement that people's attitudes about risk, or what constitutes an acceptable risk, can or should be governed solely by such a measure. At best, it can be a contributing factor to a risk judgment, but not necessarily a decisive factor.

In Table 7.5 are shown average values of risk associated with population dose, acute fatalities, latent fatalities, and costs for evacuation and other protective actions. These average values are obtained by summing the probabilities multiplied by the consequences over the entire range of the distributions. Since the probabilities are on a per-year basis, the averages shown are also on a per-year basis.

Table 7.5 Annual Average Values of Environmental Risks Due to Accidents

Population exposure	
man-rem within 80 km	170
man-rem total	380
Acute fatalities	
permanent residents	0.001
beach visitors	0.00002
Latent cancer fatalities	
all organs excluding thyroid	0.022
thyroid only	0.011
Cost of protective actions	
and decontamination	$19,000

NOTE: See Section 7.1.4.6 for discussions of uncertainties in risk estimates.

The population exposure risk due to accidents may be compared with that for normal operational releases. These are shown in Section 5, Tables 5.3 and 5.5, for San Onofre Units 2 and 3 operating concurrently. The radiological dose to the population from normal operational releases may result in:

(1) late somatic effects in the form of fatal and nonfatal cancer in various body organs--following age and organ-specific latency periods--of the exposed population, and

(2) fatal and nonfatal genetic disorders in the future generations of the exposed population.

Because of the randomness of these effects, calculations of these effects are made from the population dose (man-rem). Absolute risk estimators of 140 deaths from expression of latent cancer in various body organs per 10^6 total-body man-rem in the exposed population and 260 cases of all forms of genetic disorders per 10^6 total-body man-rem in the future generations of the exposed population were derived from the 1972 BEIR report.[5] This derivation assumes a linear and nonthreshold dose-effect relationship at all sublethal dose levels. Using these risk estimators and 228 man-rem as the annual population dose (Table 5.5, adjusted for one reactor), the staff calculated that there may occur 0.03 cancer deaths in the exposed population and 0.06 genetic disorders in all future generatons of the exposed population from each year of operation of one reactor.

The comparison of 0.03 cancer deaths given above with about the same value for latent cancer deaths from Table 7.1.4-5 shows that the accident risks are comparable to those for normal operational releases.

There are no acute fatality nor economic risks associated with protective actions and decontamination for normal releases; therefore, these risks are unique for accidents. For perspective and understanding of the meaning of the acute fatality risk of 0.001 per year, however, the staff notes that to a good approximation the population at risk is that within about 16 km (10 mi) of the plant, about 92,000 persons in the year 2000. Accidental fatalities per year for a population of this size, based upon overall averages for the United States, are approximately 20 for motor vehicle accidents, 7 from falls, 3 from drowning, 3 from burns, and 1 from firearms (ref. 5, p. 577).

As a separate item under acute fatalities in Table 7.5 is an entry of 0.00002 for "Beach visitors." As discussed in Section 7.1.3.2, the beaches near the site are heavily used for recreation. The average number of visitors has been estimated, based on seasonal and daily variation. The effects on the visitors are tallied separately because in actuality they are likely to be permanent residents from other nearby locations.

Figure 7.7 shows the calculated risk expressed as whole-body dose to an individual from early exposure as a function of the distance from the plant within the plume exposure pathway EPZ. The values are on a per-reactor-year basis and all accident sequences and release categories in Table 7.3 contributed to the dose, weighted by their associated probabilities. Calculated risk to an individual living within the plume exposure pathway EPZ of San Onofre of acute death due to potential accidents in the reactor is shown in Figure 7.8 as curves of constant risk per year to an individual as a function of distance due to potential reactor accidents. Figure 7.9 shows the same type of isopleths for death from latent cancer. Directional variation of these curves reflect the variation in the average fraction of the year the wind would be blowing into different directions from the plant. For comparison the following risk of fatality per year to an individual living in the U.S. may be noted (ref. 4, p. 577); automobile accident 2.2×10^{-4}, falls 7.7×10^{-5}, drowning 3.1×10^{-5}, burning 2.9×10^{-5}, and firearms 1.2×10^{-5}.

The economic risk associated with protective actions and decontamination could be compared with property damage costs associated with alternative energy generation technologies. The use of fossil fuels, coal or oil, for example, would emit substantial quantities of sulfur dioxide and nitrogen oxides into the atmosphere, and, among other things, lead to environmental and ecological damage through the phenomenon of acid rain (Ref. 4, 559-560). This effect has not, however, been sufficiently quantified to draw a useful comparison at this time.

There are other economic impacts and risks that can be monetized that are not included in the cost calculations discussed in Section 7.1.4.4. These are accident impacts on the facility itself that result in added costs to the public, i.e., ratepayers, taxpayers, and/or shareholders. These are costs associated with decontamination of the facility itself and costs for replacement power.

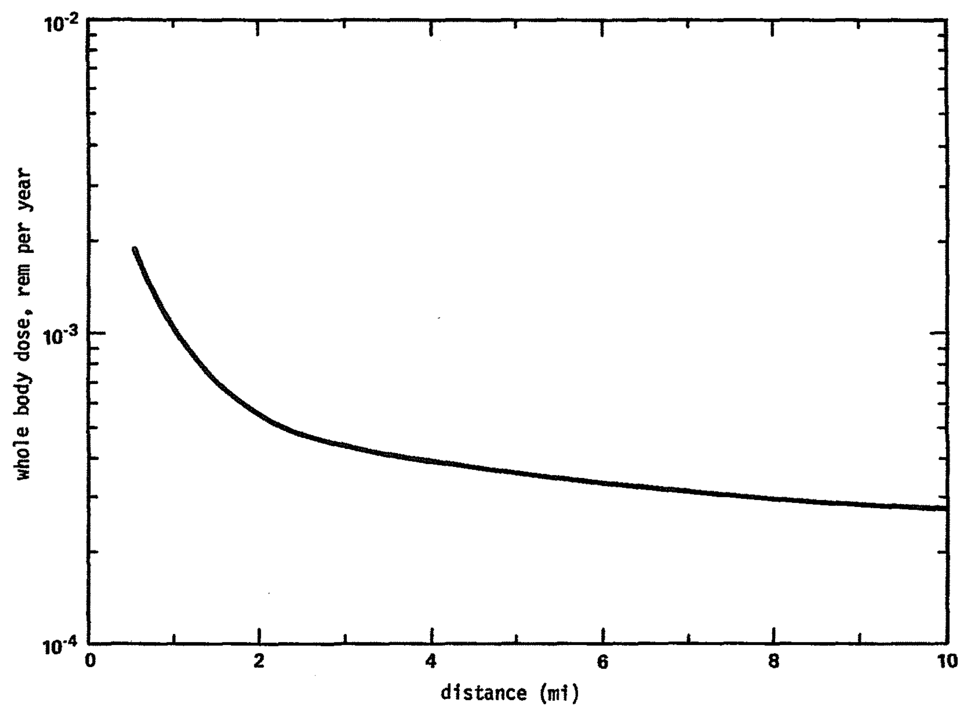


Figure 7.7 Individual risk of dose as a function of distance. (To convert mi to km, multiply by 1.6.)

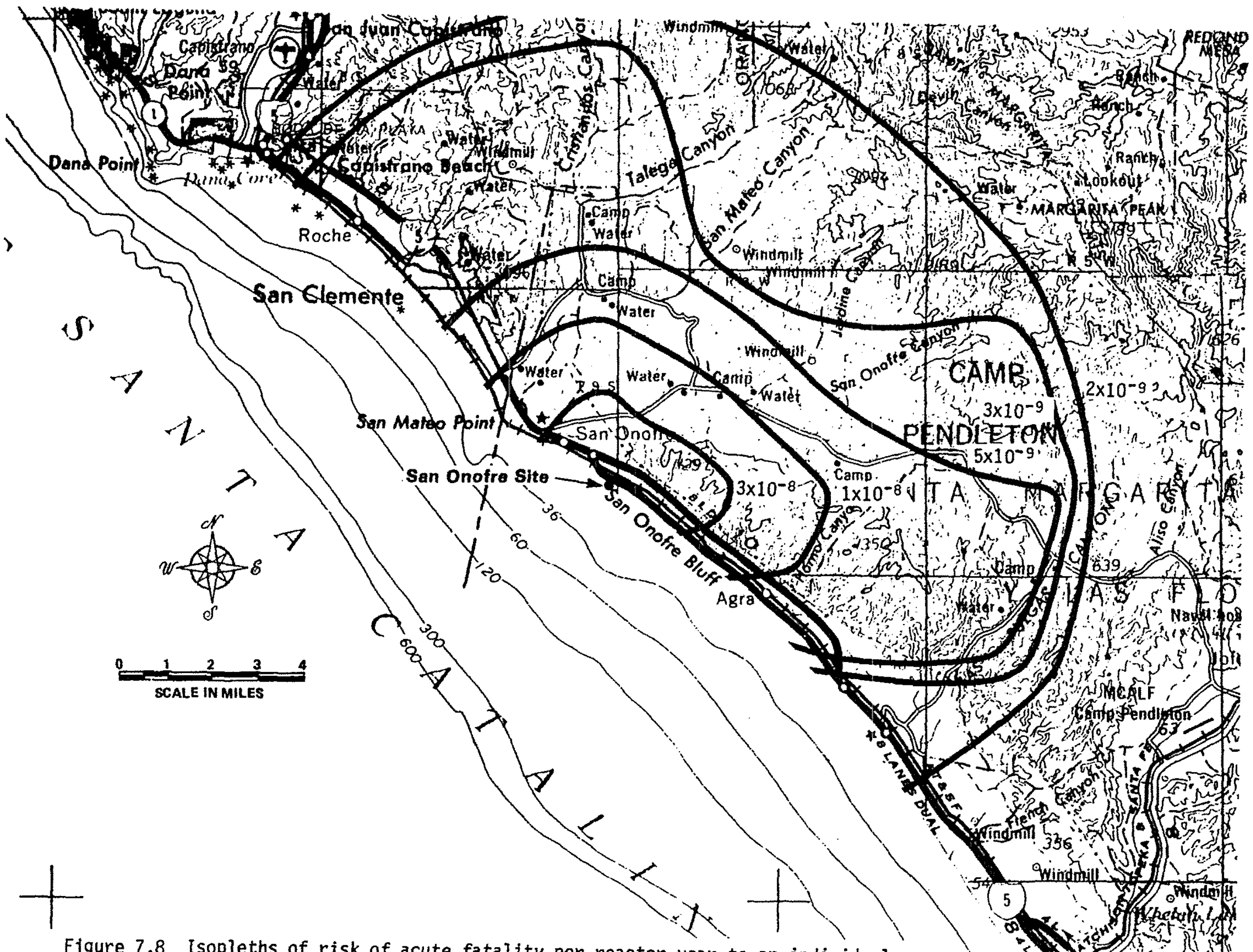


Figure 7.8 Isopleths of risk of acute fatality per reactor year to an individual, (See Section 7.1.4.6 for a discussion of uncertainties in risks estimates.) (To convert miles to kilometers, multiply by 1.6.)

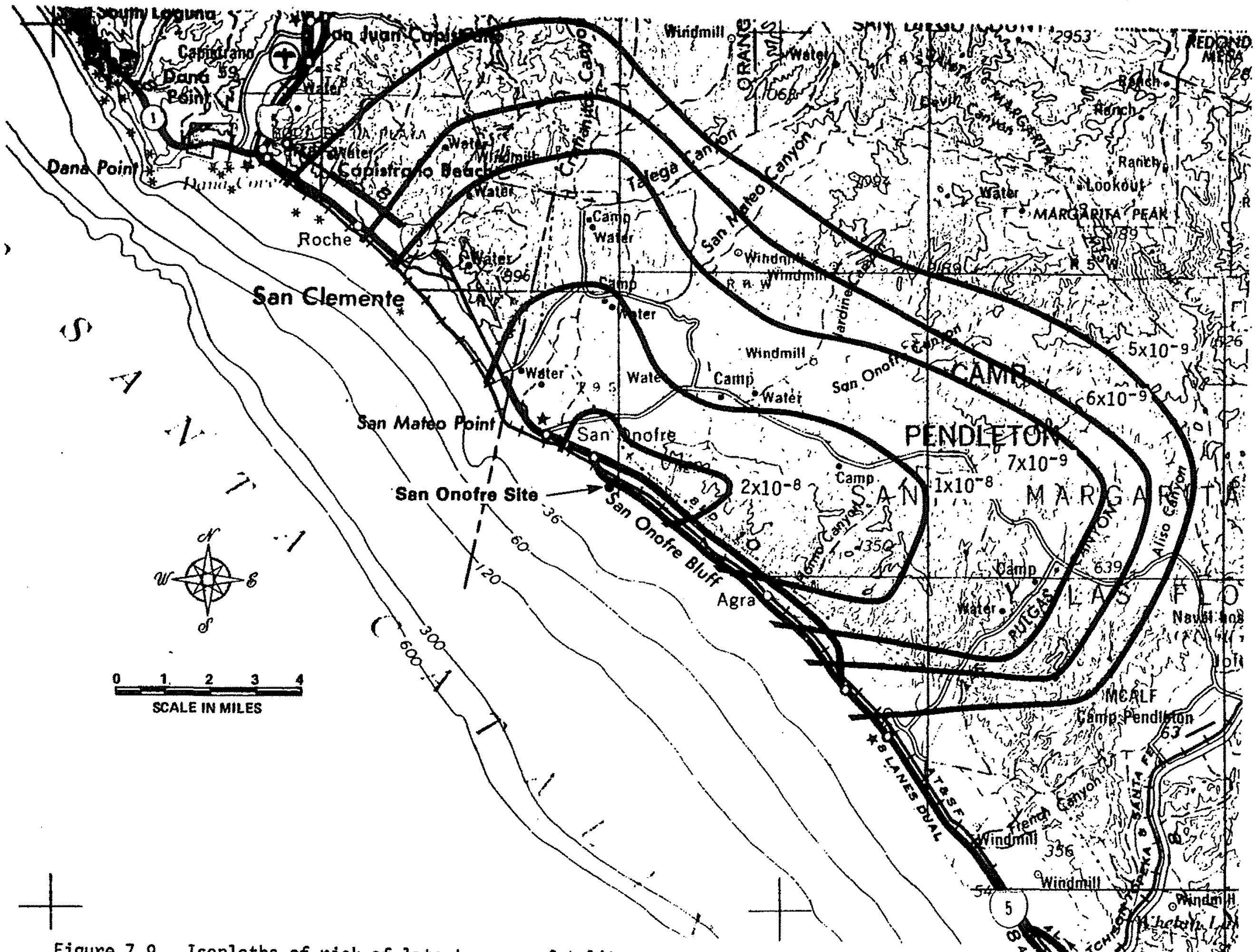


Figure 7.9 Isopleths of risk of latent cancer fatality per reactor year to an individual. (See Section 7.1.4.6 for a discussion of uncertainties in risks estimate.) (To convert miles to kilometers, multiply by 1.6.)

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No detailed methodology has been developed for estimating the contribution to economic risk associated with cleanup and decontamination of a nuclear power plant that has undergone a serious accident toward either a decommissioning or a resumption of operation. Experience with such costs is currently being accumulated as a result of the Three Mile Island accident. It is already clear, however, that such costs can approach or even exceed the original capital cost of such a facility. As an illustration of the possible contribution to the economic risk, if the probability of an accident serious enough to require extensive cleanup and decontamination is taken as the sum of the nine categories in Table 7.3, i.e., about 5 chances in 10,000 per year, and if the "average" decontamination cost for these nine categories is assumed to be one billion dollars, then the estimated economic risk would be about $500,000 per year.

Other costs, besides damage to or loss of the facility, result from accidents. The major additional costs are replacement power and replacement of the capacity. These costs are affected by the point in the lifetime of the plant at which an accident might occur. The present worth cost is highest for an accident occurring at the beginning of the plant operating life and decreases over the plant life. It is assumed for these calculations that one unit of San Onofre 2 or 3 is permanently lost and replaced by new capacity after eight years and the undamaged unit is shut down for three years before restart. For illustrative purposes, the costs and economic risk have been estimated for a "worst case" situation for the approximately 2200-megawatt (electric) San Onofre Units 2 and 3 complex by postulating a total loss of one of the units in the first year of a projected 30-year operating life. Net replacement power cost of 45 mills/kWh is assumed (nearly all fossil units in southern California are oil-fired). Using a 60% capacity factor, the annual cost of replacement power would be $520 million for the two units in 1980 dollars. The additional capital costs as a result of having to construct a new facility are $60 million per year, again in 1980 dollars.

If the probability of sustaining a total loss of the original facility is taken as the probability of the occurrence of a core melt accident (approximately by the sum of probabilities for the categories PWR-1 through 7 in Table 7.3, i.e., about 5 chances in 100,000 per year), then the average contribution to economic risk that would result from a loss early in the operating life of a San Onofre unit is about $29,000 for each of the first three years until the undamaged plant is returned to service, then $16,000 per year until the damaged unit is replaced, and $3000 per year additional capital costs for the assumed remaining 22 years of plant service.

7.1.4.7 Uncertainties

The foregoing probabilistic and risk assessment discussion has been based upon the methodology presented in the Reactor Safety Study (RSS),[10] which was published in 1975.

In July 1977, the NRC organized an Independent Risk Assessment Review Group to (1) clarify the achievements and limitations of the Reactor Safety Study Group, (2) assess the peer comments thereon and the responses to the comments, (3) study the current state of such risk assessment methodology, and (4) recommend to the Commission how and whether such methodology can be used in the regulatory and licensing process. The results of this study were issued September 1978.[11] This report, called the Lewis Report, contains several findings and recommendations concerning the RSS. Some of the more significant findings are summarized below.

(1) A number of sources of both conservatism and nonconservatism in the probability calculations in RSS were found, which were very difficult to balance. The Review Group was unable to determine whether the overall probability of a core melt given in the RSS was high or low, but they did conclude that the error bands were understated.

(2) The methodology, which was an important advance over earlier methodologies that had been applied to reactor risk, was sound.

(3) It is very difficult to follow the detailed thread of calculations through the RSS. In particular, the Executive Summary is a poor description of the contents of the report, should not be used as such, and has lent itself to misuse in the discussion of reactor risk.

On January 19, 1979, the Commission issued a statement of policy concerning the RSS and the Review Group Report. The Commission accepted the findings of the Review Group.

The accident at Three Mile Island occurred in March 1979 at a time when the accumulated experience record was about 400 reactor years. It is of interest to note that this was within the range of frequencies estimated by the RSS for an accident of this severity (ref. 4, p. 533). It should also be noted that the Three Mile Island accident has resulted in a very comprehensive evaluation of reactor accidents like that one, by a significant number of investigative groups both within NRC and outside of it. Actions to improve the safety of nuclear power plants have come out of these investigations, including those from the President's Commission on the Accident at Three Mile Island, and NRC staff investigations and task forces. A comprehensive "NRC Action Plan Developed as a Result of the TMI-2 Accident," NUREG-0660, Vol. I, May 1980 collects the various recommendations of these groups and describes them under the subject areas of: Operational Safety; Siting and Design; Emergency Preparedness and Radiation Effects; Practices and Procedures; and NRC Policy, Organization and Management. The action plan presents a sequence of actions, some already taken, that will result in a gradually increasing improvement in safety as individual actions are completed. The San Onofre plant is receiving and will receive the benefit of these actions on the schedule indicated in NUREG-0660. The improvement in safety from these actions has not been quantified, however, and the radiological risk of accidents discussed in this chapter does not reflect these improvements.

7.1.5 Conclusions

The foregoing sections consider the potential environmental impacts from accidents at the San Onofre facility. These have covered a broad spectrum of possible accidental releases of radioactive materials into the environment by atmospheric and groundwater pathways. Included in the considerations are postulated design basis accidents and more severe accident sequences that lead to a severely damaged reactor core or core melt.

The environmental impacts that have been considered include potential radiation exposures to individuals and to the population as a whole, the risk of near- and long-term adverse health effects that such exposures could entail, and the potential economic and societal consequences of accidental contamination of the environment. These impacts could be severe, but the likelihood of their occurrence is judged to be small. This conclusion is based on (a) the fact that considerable experience has been gained with the operation of similar facilities without significant degradation of the environment; and (b) a probabilistic assessment of the risk based upon the methodology developed in the Reactor Safety Study. The overall assessment of environmental risk of accidents, assuming protective action, shows that it is roughly comparable to the risk for normal operational releases although accidents have a potential for acute fatalities and economic costs that cannot arise from normal operations. The risk of acute fatalities from potential accidents at the site are small in comparison with the risk of acute fatalities from other human activities in a comparably-sized population.

The staff has concluded that there are no special or unique features about the San Onofre site and environs that would warrant special or additional engineered safety features for the San Onofre plants.

REFERENCES

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3. "Safety Evaluation Report related to the operation of San Onofre Nuclear Generating Station, Units 2 and 3 Docket Numbers 50-361 and 50-362," NUREG-0712, February 1981, as supplemented.**

4. "Energy in Transition 1985 - 2010," Final Report of the Committee on Nuclear and Alternative Energy Systems (CONAES), Chapter 9, pp. 517-534, National Research Council, 1979 (available in public technical libraries) (also C.E. Land, Science 209, 1197, September 12, 1980).

5. "The Effects on Populations of Exposure to Low Levels of Ionizing Radiation," Advisory Committee on the Biological Effects of Ionizing Radiations (BEIR), National Academy of Sciences/National Research Council November 1972.***

6. "The Effects on Populations of Exposure to Low Levels of Ionizing Radiation," Committee on the Biological Effects of Ionizing Radiations (BEIR), National Academy of Sciences/National Research Council, July 1980.***

7. H.W. Bertini and others, "Descriptions of Selected Accidents that Have Occurred at Nuclear Reactor Facilities," Nuclear Safety Information Center, Oak Ridge National Laboratory, ORNL/NSIC-176, April 1980;*** also, "Evaluation of Steam Generator Tube Rupture Accidents," L.B. Marsh, USNRC Report NUREG-0651, March 1980.**

8. "Three Mile Island - A Report to the Commissioners and the Public," Vol. I, Mitchell Rogovin, Director, Nuclear Regulatory Commission Special Inquiry Group, January 1980, Summary Section 9.*

9. "Report of the President's Commission on the Accident at Three Mile Island," October 1979, Commission Findings B, Health Effects.*

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12. "Overview of the Reactor Safety Study Consequences Model," USNRC Report NUREG-0340, October 1977.**

13. "Liquid Pathway Generic Study," USNRC Report NUREG-0440, February 1978.**

14. Dana Isherwood, "Preliminary Report on Retardation Factors and Radionuclides Migration," Lawrence Livermore Laboratories, UCID-A3.44, August 5, 1977 (available in public technical libaries).

15. San Onofre Nuclear Generating Station Units 2 and 3, Applicant's Environmental Report, Operating License Stage, Volume 2, November 1976.*

*Available for inspection and copying for a fee in the NRC Public Document Room, 1717 H St. N. W., Washington, DC 20555.
**Available from the NRC/GPO Sales Program, Washington, D. C. 20555 and the National Technical Information Service, Springfield, VA, 22161.
***Available from NTIS only.

8. NEED FOR THE STATION

8.1 RESUME

The ownership of Units 2 and 3 of the San Onofre Nuclear Generating Station is divided among Southern California Edison Company (SCE), 76.55%; San Diego Gas & Electric Company (SDG&E) 20%; the City of Riverdale, California, 1.79%; and the City of Anaheim, California, 1.66%. This section presents an analysis of the need for the station based on the energy demands of the applicant's service areas, the potential for production cost savings, and the potential for increasing the reliability of the applicant's systems.

8.2 APPLICANT'S SERVICE AREAS AND REGIONAL RELATIONSHIPS

8.2.1 Applicant's service areas

Southern California Edison Company's service area extends over a 15-county area of southern and central California, covering about 130,000 km^2 (50,000 mi^2) and containing a population in excess of 7.5 million. In 1978, SCE served 2.95 million customers, over 88% of which were residential. San Diego Gas & Electric Company supplies electricity to about 700,000 customers in San Diego County and in portions of Orange and Imperial counties. The boundaries of the service area enclose a 10,630-km^2 (4105-mi^2) area. The cities of Anaheim and Riverside serve their respective municipalities. A map of the applicant's service area is presented in Figure 8.1.

8.2.2 Regional relationships

SCE and SDG&E are members of the Western Systems Coordinating Council (WSCC) and the California Power Pool (CPP). The WSCC is the regional reliability council for the interconnected power network that serves the states west of the Rockies and parts of British Columbia. Established in 1967, the WSCC's primary function is to facilitate coordinated planning among its member systems and to provide technical support. In relation to these duties, the WSCC compiles load and resource data for the region, performs reliability studies, and recommends minimum reserve criteria. The California Power Pool, whose members are Pacific Gas & Electric Company (PG&E), SCE, and SDG&E, was formed in 1964 to provide for the continuous interconnected operation of the member utilities' power supply systems. This interconnected operation allows the utilities to make more efficient, and therefore more economical use of their generation resources and increases the overall reliability of electric service.

8.3 BENEFITS OF STATION OPERATION

8.3.1 Minimization of production costs

To minimize energy production costs, it is necessary to use the most economical mix of generation resources. The impact of the operation of SONGS 2 & 3 on the applicant's total cost of generation will be a major factor in determining the desirability of such operation. In assessing this impact, it is important to note that the fixed costs of each facility, such as the sunken capital investment and the fixed portion of the operating and maintenance costs, are irrelevant to the choice of which generation resources will be used to meet a given load, precisely because these costs are fixed and will not vary with an altered mode of system operation.

To assess the impact of station operation on the applicant's overall production costs, the staff first reviewed the latest production costs reported by the applicants for their electric generation stations. These data, presented in Tables 8.1 and 8.2, show that all oil/gas- and oil-fired facilities that are listed as base and/or intermediate units had production costs ranging from \$29.2 to \$56.7 per MWh, whereas Unit 1 of the San Onofre Nuclear Generating Station had a production cost of \$9.0/MWhr. In determining how the additional units of the San Onofre Station would compare with these figures, the staff estimated the 1983 fuel cost for these units to be \$10.8/MWhr.[1] Because SCE's and SDG&E's installed capacity is predominately oil- and oil/gas-fired, the staff concludes

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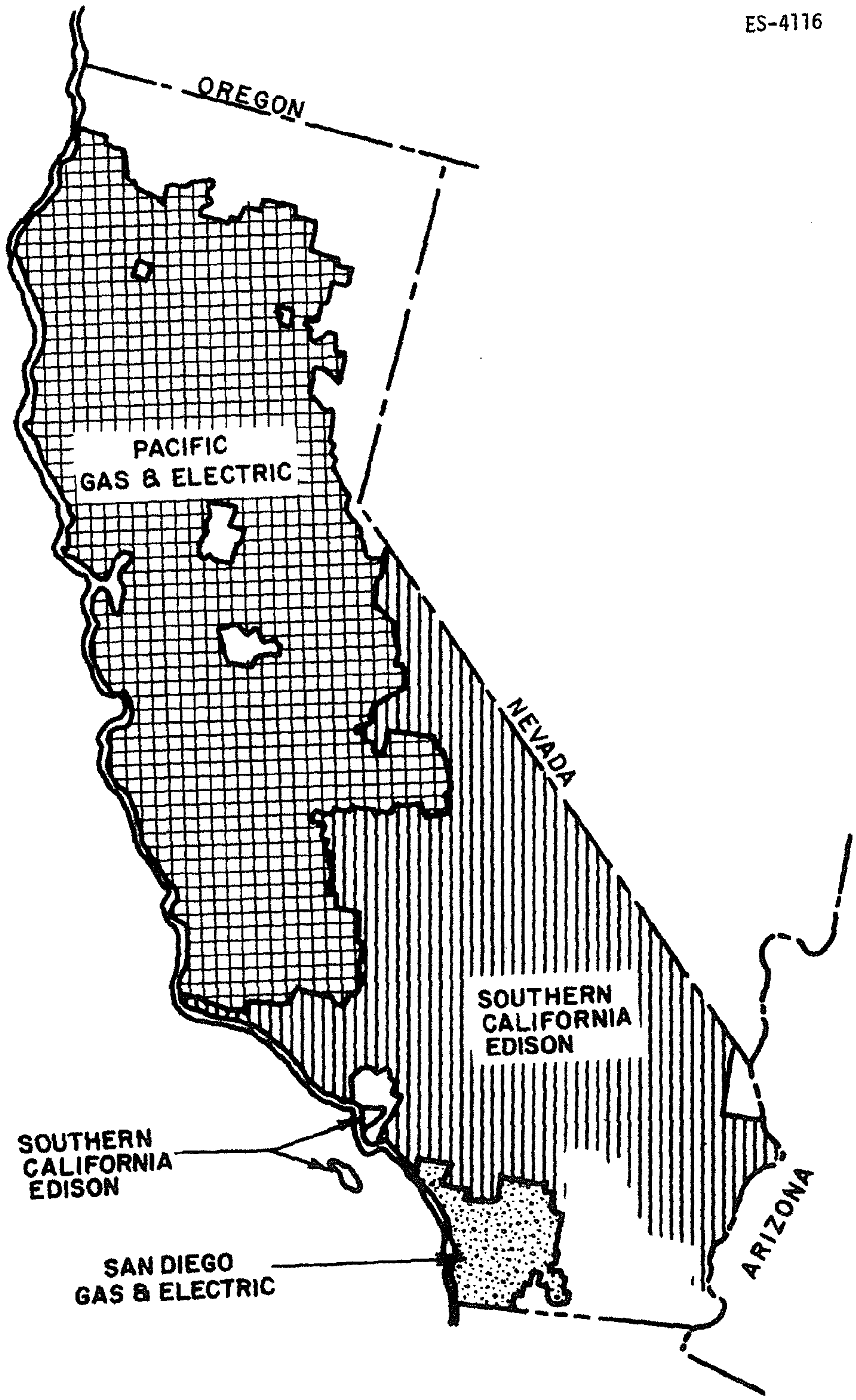

Fig. 8.1. Service areas of the member utilities of the California Power Pool. Source: ER, p. S.2-193.

Table 8.1. Southern California Edison Co. thermal-electric generating stations and production costs

Station Name	On-line dates for first and last unit	Function	Fuel type	Total station capacity (MW)	1980 Production cost (dollars/MWh)*
Long Beach	1928-1977	P	Oil/gas	156 530	77.9 38.1
Redondo Beach	1948-1967	B, I	Oil/gas	642	41.5 31.8
Huntington Beach	1958-1969	I, P	Oil/gas	870 114	39.3 49.2
Mandalay	1959-1970	I, P	Oil/gas	430 117	44.0 94.7
Ormond Beach	1971-1973	I	Oil/gas	1,500	50.6
Alamitos	1956-1969	B, I, P	Oil/gas	990 960 114	41.6 45.5 80.7
El Segundo	1955-1965	I	Oil/gas	1,020	41.8
Etiwanda	1953-1969	I, P	Oil/gas	904 108	42.2 71.8
Mohave	1971	B	Coal	885	11.8
Four Corners	1969-1970	B	Coal	768	4.6
San Onofre Unit 1	1967	B	Nuclear	349	9.0
Coolwater	1961-1978	I	Oil/gas	146 482	29.2 56.7
Highgrove	1952-1955	P	Oil/gas	154	50.5
San Bernardino	1957-1958	P	Oil/gas	126	35.0
Garden State	1967	P	Oil/gas	12	67.0
Ellwood	1974	P	Oil/gas	48	61.6

Note: B = base, I = intermediate, and P = peaking.
*Fuel only.
Source: Letter from K. P. Baskin, Southern California Edison Co., to Frank Miraglia, USNRC, undated; received by USNRC on February 25, 1981.

Table 8.2. San Diego Gas & Electric Co. thermal-electric generating stations and production costs

Station Name	On-line dates for first and last unit	Function	Fuel type	Total station capacity (MW)	1979 Production cost (dollars/MWh)
Station "B"	1923-1941	P	Oil/gas	90	188.8
Silver Gate	1943-1952	I	Oil/gas	230	48.9
Encina	1954-1978	B	Oil/gas*	917	33.7
Encina GT	1968	P	Oil/gas	16	98.6
South Bay	1960-1971	B	Oil/gas	706	33.7
South Bay GT	1966	P	Jet Fuel	18	233.4
San Onofre Unit 1	1967	B	Nuclear	87**	9.3
El Cajon	1968	P	Oil/gas	17	62.2
Kearny	1969	P	Oil/gas	147	77.5
Division	1968	P	Oil	16	97.6
Naval Training Center	1968	C	Oil/gas	16	46.0
Miramar	1972	P	Oil/gas	38	53.1
North Island	1972	P/C	Oil	41	43.7
Naval Station	1976	C	Oil/gas	26	41.0
Rohr	1979	C	Oil	1	75.8

Note: P = peaking, I = intermediate, B = base, and C = Cogeneration.
*Encina Unit 5 (320 MW) oil only.
**SDG&E's 20% share.
Source: Letter from K. P. Baskin, Southern California Edison Co., to V. A. Moore, USNRC, dated April 11, 1980.

that the operation of SONGS Units 2 and 3 would tend to reduce reliance on these facilities with corresponding savings in production costs.

To quantify the magnitude of the production cost savings, the staff made a comparison between the fuel cost that would be incurred in 1983 (the first full year in which both units are scheduled for full operation) if the two nuclear units were operated at a combined capacity factor of 50%, and the fuel cost that would be incurred if an oil-fired facility produced the same amount of electricity. In this comparison, the staff assumed a nuclear fuel cost of $10.8/MWhr in 1983, an oil cost of $4.4 per million Btu in 1983, and an oil-fired plant conversion ratio of 9,000 Btu/kWhr. These assumptions lead to an oil cost of $39.6/MWhr. All costs have been adjusted by the Producers Price Index to reflect costs in 1980 dollars. The results show that operating the nuclear units will save $270 million in fuel costs during 1983. Lowering the assumed plant capacity factor to 40% resulted in a fuel cost savings of $210 million, and raising the capacity factor to 60% gave a cost savings of $320 million. The cost of nuclear fuels would have to rise by a factor of about 3-1/2, and the price of oil would have to remain the same for the fuel savings of operating the nuclear units to disappear. These results, coupled with the information presented in Tables 8.1 and 8.2, clearly indicate that the applicant's production costs will be reduced significantly by the operation of SONGS 2 & 3.

8.3.2 Energy demand

Table 8.3 presents SCE's forecasts of peak demand, energy requirements, installed generating capacity, and reserve margins through 1985. These projections indicate that without SONGS 2 & 3 reserve margins fall below 13% from 1982-84 and dip to 7.1% in 1985. From 1980-85 SCE forecasts peak demand to grow at an average annual rate of 2.8%. A comparison with the State Level Electricity Demand[2] (SLED) forecasting model developed at Oak Ridge National Laboratory indicates that over the same period the electrical energy demand in California is forecasted to grow at an average annual rate of 4.5%. SCE's projected reserve margins without SONGS 2 & 3 clearly indicates a need for this capacity to maintain system reliability. The comparison of the applicant's forecasts of demand with the SLED forecasts reinforces the need for the additional capacity and reserve margins provided by SONGS 2 & 3.

Table 8.3. Southern California Edison Co. forecasts of peak demand, energy requirements, installed generating capacity, and reserve margins through 1985[a]

Year	Area peak demand (MW)	Growth[b] (%)	Total energy requirements kWh × 10^6	Growth[b] (%)	Installed Capacity (MW)		Reserve Margin (%)	
					With SONGS 2 & 3	Without SONGS 2 & 3	With SONGS 2 & 3	Without SONGS 2 & 3
1976	11292		59428		14071	14071	24.6	24.6
1977	11564	2.4	63345	6.6	14278	14278	23.5	23.5
1978	12159	2.9	63877	0.8	14966	14966	23.1	23.1
1979	12662	4.1	66217	3.7	15071	15071	19.0	19.0
1980	12841	1.4	65459	−1.1	15504	15504	20.7	20.7
1981	13274	3.4	67120	2.5	15471	15471	16.6	16.6
1982	13647	2.1	67910	1.2	16184	15304	18.6	12.1
1983	13895	1.8	70220	3.4	17446	15686	25.6	12.9
1984	14305	3.0	72590	3.4	17837	16077	24.7	12.4
1985	14735	3.0	75130	3.5	17535	15775	19.0	7.1

[a]Per November 18, 1980 Resource Plan.
[b]Percentage increase over previous year. 1976 through 1980 is recorded.
Source: Letter from K. P. Baskin, Southern California Edison Co., to Frank Miraglia, USNRC, undated, received by USNRC on February 25, 1981.

Table 8.4 provides analogous projections of electricity demand, installed capacity, and reserve margins for SDG&E. Without SONGS 2 & 3 reserve margins drop below 15% in 1984 and below 10% in 1985. The average annual growth in peak demand has been forecast at 1.3% which is significantly below the 4.5% rate forecast by SLED[2] for electrical energy demand in the State of California. Reserve margins forecast by SDG&E indicate a need for

Table 8.4. San Diego Gas & Electric Co. forecasts of peak demand, energy requirements, installed generating capacity, and reserve margins through 1987

Year	Area peak demand[a] (MW)	Growth[b] (%)	Energy requirements kWh × 10^6	Growth[b] (%)	Installed Capacity (MW)[c]		Reserve Margin (%)[c]	
					With SONGS 2 & 3	Without SONGS 2 & 3	With SONGS 2 & 3	Without SONGS 2 & 3
1978[d]	1981	13.5	10053	7.8	2030	2030	2.5	2.5
1979[d]	2019	1.9	10548	4.9	2363	2363	17.0	17.0
1980[d]	2050	3.7	10403	-1.4	2401[e]	2401[e]	17.1	17.1
1981	1975	-3.7	10738	3.2	2366	2366	19.8	19.8
1982	2004	1.5	10824	0.8	2586	2366	29.0	18.1
1983	2033	1.4	11108	2.6	2806	2366	38.0	16.4
1984	2077	2.2	11407	2.7	2806	2366	35.1	13.9
1985	2184	5.2	11762	3.1	2806	2366	28.5	8.3
1986	2272	4.0	12244	4.1	2806	2366	23.5	4.1
1987	2361	3.9	12763	4.2	2806	2366	18.8	0.0

[a] 1981–1987 SDG&E CFM III Forecast adopted by California Energy Commission in December 1980.
[b] Percentage increase over previous year.
[c] Excludes purchased capacity.
[d] 1978 through 1980 are recorded.
[e] July net rating.
Source: Letter from K. P. Baskin, Southern California Edison Co., to Frank Miraglia, USNRC, undated, received by USNRC on February 25, 1981.

SONGS 2 & 3 to maintain system reliability. Once again comparing the applicant's forecasts to the SLED forecasts reinforces the need for the additional capacity and reserve margins provided by SONGS 2 & 3.

The staff concludes on the basis of the analysis of the applicant's projected reserve margins that operation of SONGS 2 & 3 will be needed to ensure reliability within the time frame that operation is anticipated to begin. Furthermore, the analysis of cost savings due to a shift from oil-fired to nuclear generation (Sect. 8.3.1) makes operation of SONGS 2 & 3 economically desirable independent of load forecasts.

REFERENCES

The documents references below are available from the NRC/GPO Sales Program, Washington, DC 20555 and the National Technical Information Service, Springfield, VA 22161.

1. J. O. Roberts, S. M. Davis, and D. A. Nash, "Coal and Nuclear: A Comparison of the Cost of Generating Baseload Electricity by Region," U.S. Nuclear Regulatory Commission Report NUREG-0480, December 1978.

2. W. S. Chern, J. W. Dick, C. A. Gallagher, B. D. Holcomb, R. E. Just, and H. D. Nguyen, "The ORNL State-Level Electricity Demand Forecasting Model," Oak Ridge National Laboratory Report ORNL/NUREG-63, July 1980.

9. CONSEQUENCES OF THE PROPOSED ACTION

9.1 ADVERSE EFFECTS THAT CANNOT BE AVOIDED

The staff has reassessed the physical, social, and economic impacts that can be attributed to SONGS 2 & 3. The identification of adverse effects that cannot be avoided, given in Chap. 8 of the FES-CP, remains valid. The major effects identified were the destruction of a small amount of wildlife habitat in the area occupied by the plant buildings and the loss of fish and other marine organisms that will be entrained in the circulating cooling water system. In addition, construction has resulted in the excavation of about 16.4 ha (40.5 acres) of the San Onofre Bluffs, and operation of the plant will result in the removal of approximately 1.4 km (0.85 mile) of beach from unrestricted public use.

9.2 SHORT-TERM USES AND LONG-TERM PRODUCTIVITY

The assessment of the short-term uses and long-term productivity contained in Chap. 9 of the FES-CP remains valid. About 21 ha (52 acres) of the total of 36 ha (90 acres) comprising all three units will be devoted to the production of electrical energy for the next 30 to 40 years. If, at the end of this period, the site is no longer needed for the production of electrical energy, it could be used for other purposes (see Sect. 9.4, below).

9.3 IRREVERSIBLE AND IRRETRIEVABLE COMMITMENT OF RESOURCES

There has been no change in the staff's assessment of these commitments since the earlier review (FES-CP, Chap. 10) except that the continuing escalation of costs has increased the dollar values of the materials used for construction and for fueling the plant. The staff has, however, expanded and updated the discussion on uranium fuel availability. This updated discussion is presented below.

9.3.1 Replaceable components and consumable materials

Uranium is the principal natural resource irretrievably consumed in facility operation. Other materials consumed, for practical purposes, are fuel-cladding materials, reactor-control elements, other replaceable reactor core components, chemicals used in processes such as water treatment and ion-exchanger regeneration, ion-exchange resins, and minor quantities of materials used in maintenance and operation. Except for the uranium isotopes U-235 and U-238, the consumed resource materials have widespread use; therefore, their use in the proposed operation must be reasonable with respect to needs in other industries. The major use of the natural isotopes of uranium is for production of useful energy.[1]

The reactor will be fueled with uranium enriched in the isotope U-235. After use in the plant, the fuel elements will still contain U-235 slightly above the natural fraction. This slightly enriched uranium, if separated from plutonium and other radioactive materials (separation would take place in a chemical reprocessing plant), would be available for recycling through the gaseous diffusion plant if required. Scrap material containing valuable quantities of uranium may also be recycled through appropriate steps in the fuel production process. Should chemical reprocessing of spent fuel be effected in the future, the fissionable plutonium recovered is potentially valuable for fuel in power reactors.

In view of the quantities of materials in natural reserves, resources, and stockpile and the quantities produced yearly, the expenditure of such material for the power facility is justified by the benefits from the electrical energy produced. A detailed discussion of uranium supply and demand follows.

9.3.2 Uranium resource availability

This section reviews information available from the Department of Energy (DOE) on the domestic uranium resource situation and the outlook for development of additional domestic supplies, availability of foreign uranium, and the relationship of uranium supply to planned nuclear generating capacity.

Analysis of uranium resources and their availability has been carried out by the government since the late 1940's. The work was carried out for many years by the Atomic Energy Commission (AEC). The activity was made part of the Energy Research and Development Administration (ERDA) when the agency was created in early 1975[1] and was subsequently transferred to the DOE when it was formed October 1, 1977.

9.3.2.1 U.S. resource position

To establish some basic terminology, a review of resource concepts and nomenclature would be worthwhile. Figure 9.1 defines resource categories based on varying geologic knowledge. Resources designated as ore reserves have the highest assurance regarding their magnitude and economic availability. Estimates of reserves are based on detailed sampling data, primarily from gamma ray logs of drill holes. DOE obtains basic data from industry from its exploration effort and estimates the reserves in individual deposits. In estimating ore reserves, detailed studies of feasible mining, transportation, and milling techniques and costs are made. Consistent engineering, geologic and economic criteria are employed. The methods used are the result of over 30 years of effort in uranium resource evaluation.

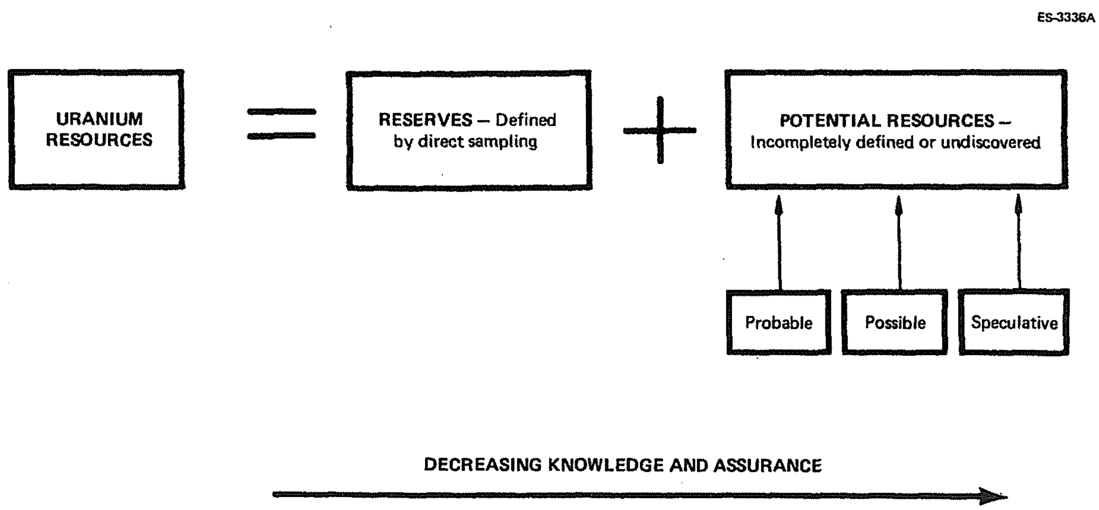

Fig. 9.1. DOE uranium resource categories.

Resources that do not meet the stringent requirements of reserves are classed as potential resources. For its study of resources, DOE subdivides potential resources into three categories: probable, possible, and speculative.[2] Probable potential resources are those contained within favorable trends, largely delineated by drilling, within productive uranium districts, i.e., those having more than 10 tons of U_3O_8 production and reserves. Quantitative estimates of potential resources are made by considering the extent of the identified favorable areas and by comparing certain geologic characteristics with those associated with known ore deposits.

Possible potential resources are outside of identified mineral trends but are in geologic provinces and formations that have been productive. Speculative resources are those estimated to occur in formations or geologic provinces which have not been productive but which, based on the evaluation of available geologic data, are considered to be favorable for the occurrence of uranium deposits.

Because any evaluation of resources is dependent upon the availability of information, the estimates themselves are, to a large degree, a scorecard on the state of development of information. Thus, appraisal of U.S. uranium resources is heavily dependent on the completeness of exploration efforts and the availability of subsurface geologic data. Since the geology of the United States as it relates to mineral deposits can never be completely known in detail, it will not be possible to produce a truly complete appraisal of domestic uranium resources. It is likely that the total resource picture will eventually prove larger than currently estimated given the nature and status of estimation methodology. The key factor may be the timeliness with which resources are identified, developed, and produced.

Conceptually, a resource, whether uranium or other mineral commodity, would initially be in the potential category. Development of additional data and clarification of production techniques and economics would be required to delineate and understand specific ore deposits to a degree that they could be categorized as reserves.

We can expect a dynamic balance between anticipated markets and prices and the extent to which exploration and reserve delineation will be done. There is no economic incentive for industry to expand reserves if the additional uranium will not be needed for many years ahead, and especially if the long-term market outlook is uncertain. This has been true for uranium. The mining companies are concentrating on markets for the next 5 to 15 years. The utilities and government are concerned with the outlook for the next 30 to 40 years.

Conversion of the currently estimated potential resources into ore reserves will take many years and will cost several billion dollars. It would be difficult to economically justify accelerating such an effort to delineate ore reserve levels equal to lifetime requirements of all planned reactors covering some 30 to 40 years in the future simply to satisfy planners. Supply assurance through continued timely additions to reserves and maintenance of a resource base adequate to support production demands, coupled with carefully developed information on potential resources, is considered to be adequate and a more realistic and economic approach. The conversion of potential resources to ore reserves and expansion of production facilities can be accomplished when needed as markets expand and production is needed.

All uranium resource estimates made by DOE and its predecessor agencies before 1979 were single estimates of tons of ore and grade for various cost categories. The estimtes were made by experienced geologists and engineers according to standard procedures, and represented a reasonable measure of resources. The current procedures for estimating uranium resources provide both mean values and distributions to characterize the reliability of the estimates at specific confidence levels. All available geologic information and the expertise of the estimators are fully utilized. These procedures are standardized and documented to minimize personal biases and to facilitate reviews and revisions as new information is acquired.

The estimates of resources in the United States are developed from a data base accumulated during the past three decades of Government and industry activities and enhanced by National Uranium Resource Evaluation program investigations of the past five years. Data acquired to support resource assessment have been extensive and varied. The assessment includes the evaluation of several hundred thousand industry-drilled holes; aerial radiometric surveys; sampling and geochemical analyses of groundwater, stream water, and stream sediment; selective drilling to fill voids in subsurface information; and extensive geologic field examinations. These data have been evaluated to determine those areas favorable for uranium occurrences. Evaluation criteria have been developed from studies of uranium deposits throughout the world. In favorable areas, the uranium endowment, material greater than 0.01 percent U_3O_8, is estimated, and subsequently economic factors are applied to assess the potential resources available at selected costs.

The costs used to calculate uranium resources are forward costs which consider both operating and capital costs, in current dollars, that would be incurred in producing the uranium. These costs include power, labor, materials, royalties, payroll, severance and ad valorem taxes, insurance, and applicable general and administrative costs. All previous expenditures (before the time of the estimate) for such items as property acquisition, exploration, mine development, and mill construction are excluded. Also excluded are income taxes, profit, and the cost of money. The resources assigned to the various cost categories are independent of the market price at which the uranium might be sold.

There are two major methodologies in uranium assessment; one is used for the estimation of reserves based on sample results from drill holes on specific properties; the second involves the use of a variety of geologic information to subjectively estimate potential resources. Reserves are calculated individually for properties throughout the United States using data voluntarily provided by the uranium companies to DOE. The data consist primarily of radiometric drill hole logs and maps. Parameters evaluated include thickness and tenor of mineralized rock; depth and spatial relationships, mining methods, ore dilution, and recovery; and amenability of ores to processing. The amounts of uranium that could be exploited at the forward cost levels are calculated according to conventional engineering practices utilizing available engineering, geologic, and economic data.

A regional reserves distribution estimate is obtained by mathematically combining the estimates of individual distributions for each property. These regional distributions are then combined to provide a total for the United States. Estimates include all material over a selected minimum thickness with a uranium content above 0.01% U_3O_8. A recovery factor is applied, after rate procedures are used for properties on which solution mining is in progress or is planned.

Potential resource estimates are based on geologic analogy. Geologic characteristics related to uranium potential in the area being investigated are compared with those in an area with similar characteristics, that is, a control area that contains uranium deposits for which the frequency distribution of grades and tonnages in the deposits has been developed. The analogy-based methodology is made feasible by DOE's extensive data base from which detailed characterizations of the distribution of uranium have been developed. From systematic comparison with an appropriate control area, an estimate is developed of the total amount of uranium, above 0.01% U_3O_8, that might be present in an area being evaluated. Uranium endowment factors, such as surface area, fraction underlain by endowment, grade, and tonnage are estimated at three confidence levels, i.e., a modal value which is considered as most likely, and a low and high estimate corresponding respectively to a 95 and 5% probability that the factor is at least that large. The endowment estimate is analyzed to determine the portions that are producible at various cost categories within stated confidence levels.

Table 9.1 provides the mean reserve and potential resource estimates for each cost category, as well as estimates at the 95th and 5th percentile. The 95th percentile value provides an estimate for which there is a 95% confidence that at least that amount exists. The 5th percentile provides an estimate for which there is a 5% probability that it will be exceeded. Due to the correlation of the individual estimates that are aggregated to generate the regional and national totals, the estimates at the 95th and 5th percentile are not directly additive; however, the mean values are additive.

Table 9.1. Uranium resources of the United States[a]

Reserves as of January 1, 1980
Other Resources as of October 1, 1980
Tons U_3O_8
Probability distribution values

Forward-cost category	Mean	95th percentile	5th percentile
At $15 per pound of U_3O_8[c]			
Reserves	225,000	190,000	260,000
Probable	295,000	185,000	448,000
Possible	87,000	42,000	156,000
Speculative	74,000	30,000	162,000
Totals	681,000	447,000	1,026,000
At $30 per pound of U_3O_8[b,d]			
Reserves	645,000	567,000	729,000
Probable	885,000	659,000	1,161,000
Possible	346,000	194,000	530,000
Speculative	311,000	155,000	600,000
Totals	2,187,000	1,731,000	2,748,000
At $50 per pound of U_3O_8[b,e]			
Reserves	936,000	821,000	1,060,000
Probable	1,426,000	1,102,000	1,802,000
Possible	641,000	346,000	973,000
Speculative	482,000	251,000	890,000
Totals	3,485,000	2,771,000	4,313,000
At $100 per pound of U_3O_8[b,f]			
Reserves	1,122,000	971,000	1,291,000
Probable	2,080,000	1,646,000	2,573,000
Possible	1,005,000	521,000	1,526,000
Speculative	696,000	378,000	1,225,000
Totals	4,903,000	3,875,000	6,056,000

[a]Uranium resources are estimated quantities recoverable by mining.
[b]Includes lower cost resource categories.
[c]$6.80 per kilogram.
[d]$13.60 per kilogram.
[e]$22.65 per kilogram.
[f]$45.30 per kilogram.
(To convert pounds to kilograms, multiply by 0.454; to convert tons to tonnes, multiply by 0.907.)

Most of the uranium resources are located in a few areas in the Colorado Plateau of New Mexico, Arizona, Colorado, and Utah, in the Wyoming Basins, and in the Texas Gulf Coastal Plain, Figs. 9.2 and 9.3. It should be noted that the reserve estimates in Table 9.1 were as of January 1, 1980, and the lower cost reserves have undoubtedly decreased since that date because of continuing rising costs.

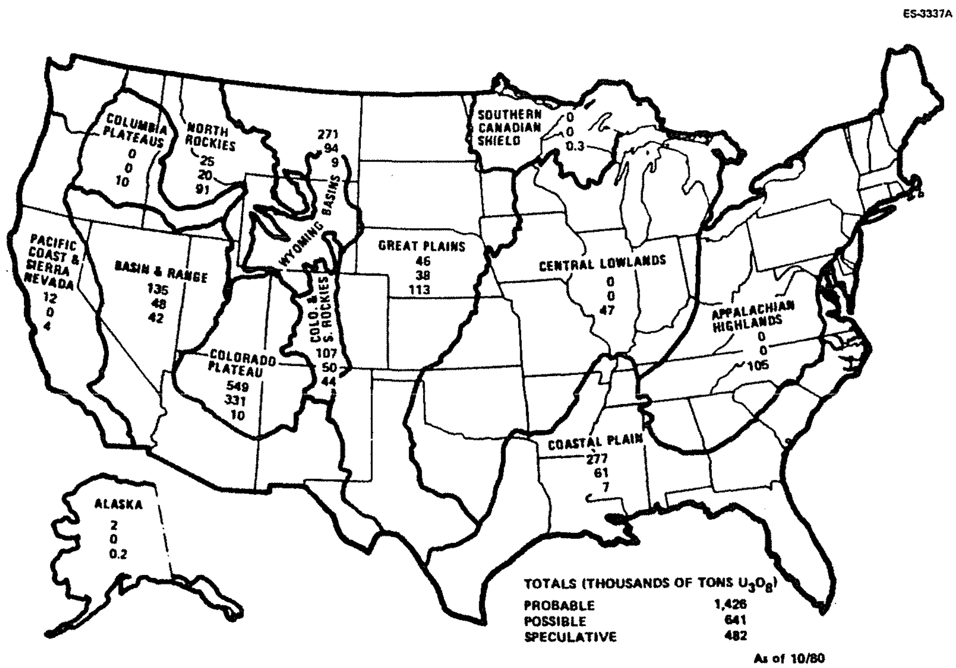

Fig. 9.2. Potential uranium resources by region ($22.65 per kilogram ($50 per pound) of U_3O_8).

9.3.2.2 Uranium exploration activities

Uranium exploration in the United States reached its all time high in 1978 as measured by the principal exploration indicator, surface drilling. Data provided to DOE by the exploration companies indicated a total of 14.6 million meters (48 million feet) of drilling in 1978. In 1979, however, drilling declined to 12.5 million meters (41 million feet), and during 1980 the downward trend steepened with drilling estimated to be approximately 8.5 million meters (28 million feet) for the year (see Figure 9.4).

Annual gross additions to reserves, a measure of exploration success, have been at high levels for the higher cost, i.e., $13.60 to $22.65 per kilogram ($30 and $50 per pound) U_3O_8 categories, but have been decreasing for lower cost levels. Costs have increased significantly in recent years raising the quality of resources needed to produce at a given cost level and reducing the quantities available at that level. For example, in 1979 only 907 tonnes (1000 tons) were added to $6.80 ($15) cost revenues, but 47,164 tonnes (52,000 tons) were removed, largely because of inflation, and an additional 12,698 tonnes (14,000 tons) were depleted by production. Hence, in 1979, $6.80 ($15) reserves decreased from 263,030 to 204,075 tonnes (290,000 to 225,000 tons). This trend continued in 1980. On the other hand, in 1979 some 84,351 tonnes (93,000 tons) were added to $22.65 ($50) reserves and 69,839 tonnes (77,000 tons) removed for a net increase of 14,512 tonnes (16,000 tons) U_3O_8. Thus, while exploration has been successful, the costs of producing the resources found are high in comparison with current prices and concurrently the cost of producing previously found resources has also increased.

The sharp rise in exploration resulted from the increase in prices in the 1974 to 1976 period, the active procurement activity of utilities, and the optimistic projections of future growth in uranium demand. Many new companies became active in exploration. Over 150 companies were involved in exploration in 1979. Considering the drop in requirement projections the level of activity reached probably was in excess of real needs. Therefore, some reduction of effort more in line with future needs is not detrimental.

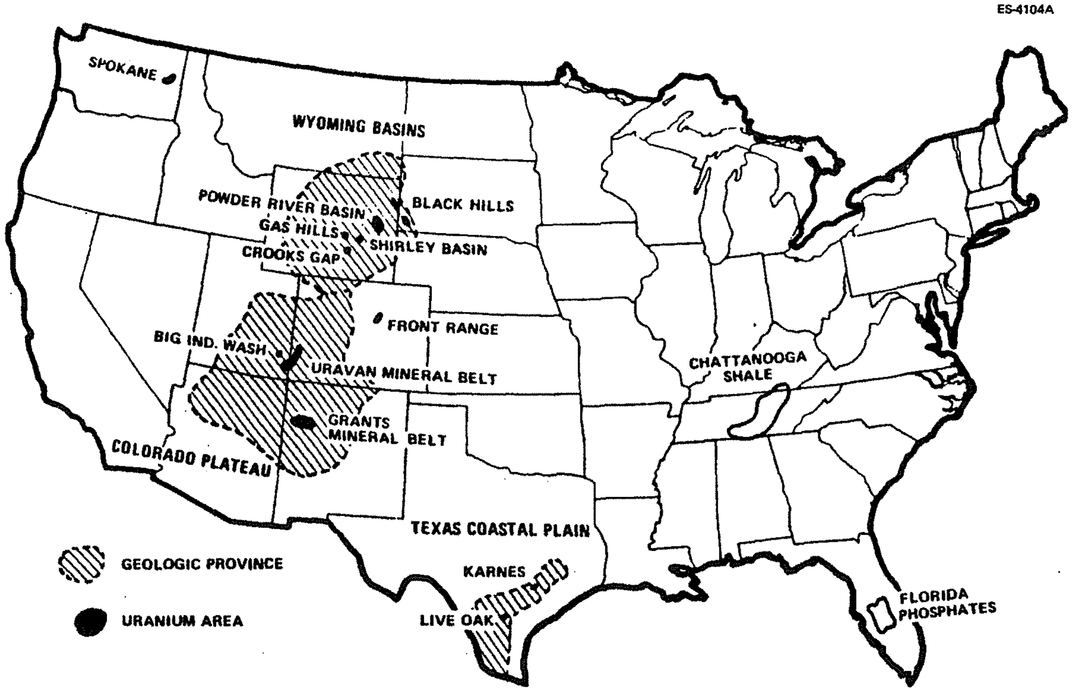

Fig. 9.3. Uranium areas of the United States.

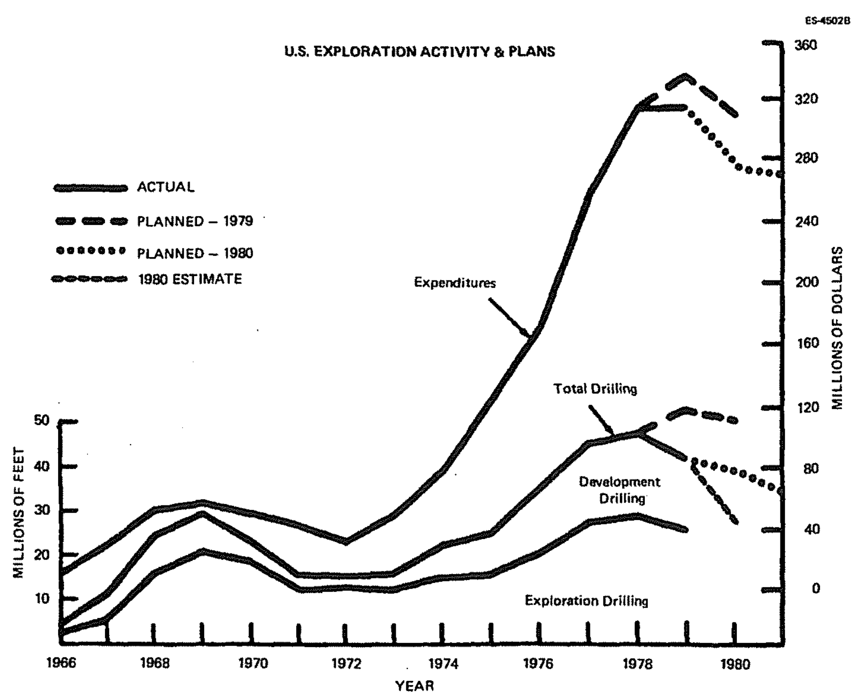

Fig. 9.4. U.S. exploration activity and plans. (To convert feet to meters, multiply by 0.3048.)

9.3.2.3 Domestic uranium production and capability

Domestic uranium production in 1980 was 19,573 tonnes (21,850 tons) U_3O_8 in concentrate. This represents a 15% increase over 1979 and is the highest U.S. production level for any single year. Production in recent months has been at record rates; the equivalent of over 19,954 tonnes (22,000 tons) U_3O_8 per year. This production comes from conventional mine-mill operations as well as nonconventional sources such as solution mining and byproduct recovery from processing of other minerals. The high production levels are in response to prior sales contracts. Buyers are actually receiving uranium in excess of their currently scheduled needs.

Several new uranium processing facilities are under construction or planned which could bring the total national capacity to around 27,000 tonnes (30,000 tons) per year by the mid-1980s.

Despite the increases in ore throughput and uranium production in 1980, a widespread curtailment of uranium mining and milling activities is underway. Production at some operating mines has been reduced and some planned mill expansions and construction are being postponed. The reduction in mine output will not be reflected in decreased uranium production until mine and mill ore stockpiles are reduced.

Studies have been conducted on attainable uranium production levels from uranium reserves in the United States and related costs. The uranium production capability projections should not be construed as being estimates of actual future supply, but simply as potential production which may be available to meet whatever demand eventually exists.

Using the "production center" concept, U.S. uranium production capability has been projected from ore reserves estimated as of January 1980, to be available at forward costs of $13.60 to $22.65 per kilogram ($30 and $50 per pound) U_3O_8 or less. The production centers consist of operating (Class 1), committed (Class 2), planned (Class 3) uranium extraction and processing facilities, and projected (Class 4) facilities based on probable potential resources. The study included conventional mills supplied by open pit and/or underground mines; solution mining and heap-leach operations; and operations where uranium is recovered as a byproduct of phosphate, copper, or beryllium mining and processing activities.

Projections are based primarily on operating conditions – average ore grades, mill recoveries, and operating and capital costs – similar to those currently prevalent in the uranium mining and milling industry. Specific information on company plans, costs, and operating methods has been considered.

Figure 9.5 shows the total projected production capability for $13.60 ($30) resources by resource category. Figure 9.6 shows the capability for $22.65 ($50) resources. Projected uranium demand and current sales commitments are also shown. Domestic demand is based on the DOE's Office of Uranium Resources and Enrichment 1980 nuclear power growth projections, assuming no reprocessing and a 0.20% U-235 enrichment tails assay.

9.3.2.4 Domestic reactor requirements

The outlook for uranium requirements is closely related to the growth of nuclear power. On December 1, 1980, there were 75 nuclear power reactors licensed to operate in the U.S., concentrated mostly in the East and Midwest. These plants have an electrical generating capacity of 55 Gigawatts (GWe). In addition to operating plants, there are under construction 86 plants with a total rated capacity of 95 GWe. Some of the plants are at such an early construction stage that they may be deferred or cancelled completely. An additional 17 reactors with 20 GWe capacity are on order. Together the group aggregates 170 GWe of capacity. However, the future for some of the ordered reactors is questionable.

Latest projections of nuclear power growth by the DOE's Office of Uranium Resources and Enrichment (URE) and the Energy Information Administration (EIA), Table 9.2, show an increase in nuclear power licensed to operate from 55 GWe at the end of 1980 to 96 GWe in 1985, 129 GWe in 1990, 155 GWe in 1995, and 180 GWe in 2000. EIA also projected a low case of 160 GWe and a high case of 200 GWe in 2000.

There are alternative views on U.S. power growth. The DOE's Office of Planning and Analysis has projected nuclear growth to the year 1990 at 125 GWe and to the year 2000 at 150 GWe, based on historic delays to nuclear power growth. The DOE Office of the Assistant Secretary of Nuclear Energy has projected 400 GWe, based on energy demand, growth, nuclear competitiveness, and industry construction capability. All of these values are sharply reduced from the projected growth of the nuclear industry of just a few years ago. For example, in 1976 U.S. nuclear capacity in the year 2000 had been projected to be 500 GWe, and in 1978 it had been projected to be 320 GWe.

ES-5733

ESTIMATED ANNUAL NEAR-TERM PRODUCTION CAPABILITY FROM RESOURCES AVAILABLE AT $30/LB U_3O_8 OR LESS WITH CLASS 1, 2, AND 3 EXPANSIONS AND CLASS 4

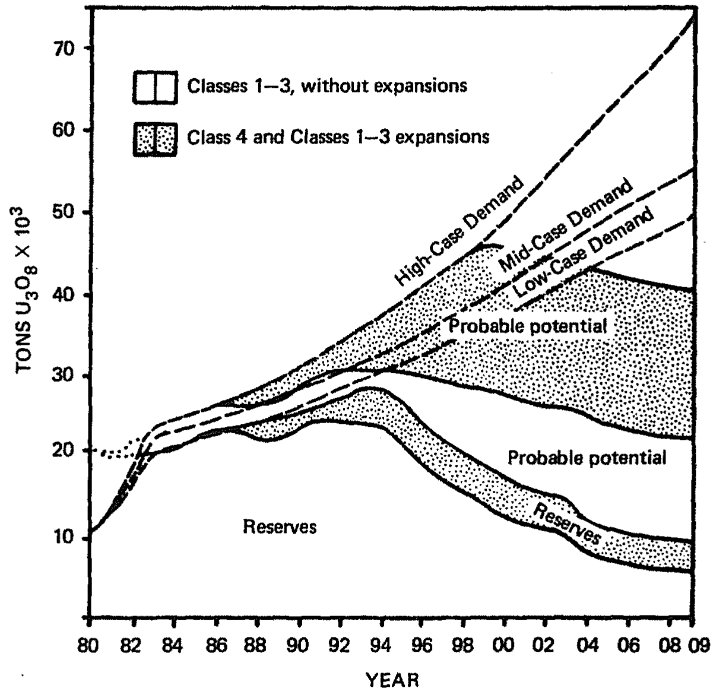

Fig. 9.5. Estimated annual near-term production capability from resources available at $13.60 per kilogram ($30 per pound) of U_3O_8 or less with Class 1, 2, and 3 expansions and Class 4. (To convert tons to tonnes, multiply by 0.907.)

Even at the more conservative estimates, nuclear capacity still is expected to expand substantially and to provide a significant portion of future domestic electric capacity. Current methods of projecting nuclear growth and uranium requirements are based on estimates of reactor startup dates considering construction and licensing times, and systems power requirements. Accurate forecasts have proven to be difficult.

The uranium needed to be delivered by uranium concentrate-producing plants as fuel for the nuclear plants will also increase over time; for the URE mid-case, from 12,063 tonnes (13,300 tons) U_3O_8 in 1981 to 21,405 (23,600) in 1985, 26,212 (28,900) in 1990, 31,745 tonnes (35,000 tons) in 1995, and 36,280 tonnes (40,000 tons) in 2000, if the enrichment plants are operated at 0.20% U-235 tails assay. Cumulative uranium requirements through the year 2000 range from 462,570 to 562,340 tonnes (510,000 to 620,000) tons U_3O_8 with 516,990 tonnes (570,000 tons) U_3O_8 for the mid-case.

Uranium requirements are based on normal lead times for fuel cycle steps and current technology for enrichment and for reactor design and operation. There are possible improvements in enrichment which would allow use of lower tails assays which would reduce uranium requirements. There are also possible improvements to reactor design and operation that could reduce uranium requirements. These factors are not likely to have a significant impact on uranium demands until at least well into the 1990s.

9.3.2.5 Uranium inventories

Buyers' inventories of uranium have been increasing for several years as actual deliveries have been in excess of needs. Inventories at the beginning of 1980 totalled 32,742 tonnes (36,100 tons) of natural uranium (Table 9.3), with 25,033 tonnes (27,600 tons) held by utilities. In 1980, U.S. utilities sent an

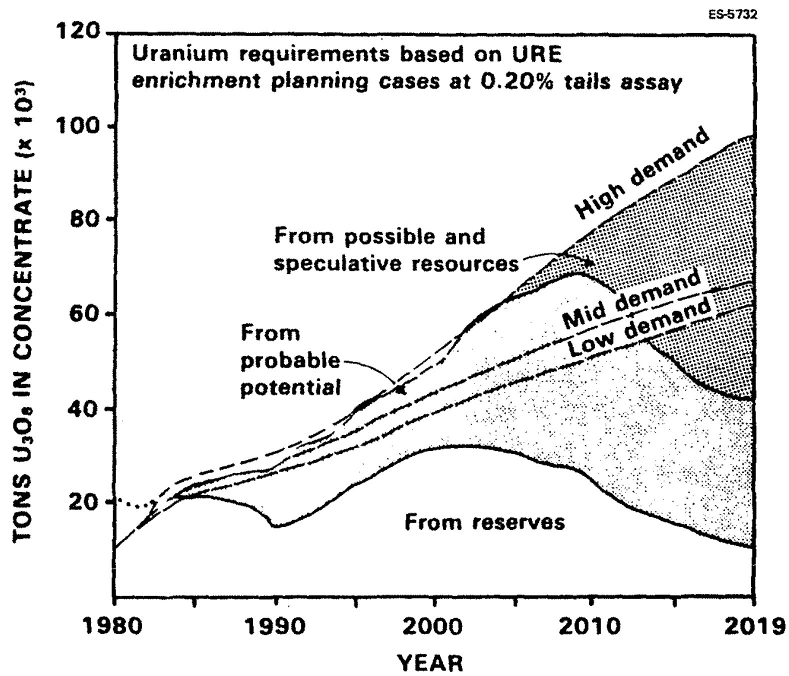

Fig. 9.6. Annual production capability from resources available at $22.65 per kilogram ($50 per pound) of U_3O_8 or less projected to meet nuclear power growth demand. (To convert tons to tonnes, multiply by 0.907.)

Table 9.2. U.S. nuclear power growth projections (June 1980)

End of year	Power Range [GW(e)] Low	Mid	High
1985	85	96	105
1990	125	129	140
1995	142	155	165
2000	160	180	200

Table 9.3. Buyers' inventories of natural uranium (Tons U_3O_8)

Beginning of year	Domestic origin	Foreign origin	Total
1976	22,600	1,100	23,700
1977	25,800	3,500	29,300
1978	25,100	3,600	28,700
1979	28,000	5,200	33,200
1980	30,800	5,300	36,100

(To convert tons to tonnes, multiply by 0.907.)

equivalent of 15,691 tonnes (17,300 tons) U_3O_8 to the DOE gaseous diffusion plants for enrichment. Thus, the 25,033 tonnes (27,600 tons) inventory level amounted to 1.6 years of U.S. utilities' needs. Of those U.S. utilities who have responded to questions on inventory levels, most have indicated that they desire a level amounting to about one year's needs, although some have reported inventory levels as small as three month's needs, while others desire inventories as great as two year's needs. Producers also had inventories of about 2,177 tonnes (2,400 tons) U_3O_8 at the beginning of 1980, which is about a normal working inventory. The outlook is for a continuing buildup of buyers' inventories, as current contracted deliveries are in excess of actual needs.

9.3.2.6 Analysis of production capability and reactor capacity

Study of attainable production capability from currently estimated $13.60 ($30) U.S. ore reserves and probable potential resource, previously referenced, indicates that production levels of 40,815 tonnes (45,000 tons) U_3O_8 per year can be achieved with aggressive resource development and exploitation including both mining and milling. Although the level may be achieved by use of domestic $13.60 ($30) ore reserves and probable resources alone, development and utilization of $30 possible and speculative categories and use of $22.65 ($50) ore reserves and potential resources would provide added assurance that the levels could be attained and sustained. Considering the use of $22.65 ($50) resource, a level of 54,240 tonnes (60,000-tons)/year supply is achievable from currently estimated resources. Such a level could be reached by the early 1990s. Imported uranium and inventories would add to the supply from these projections.

The level of nuclear generating capacity supportable with 54,240 tonnes (60,000 tons)/year of uranium, will vary with enrichment tails assay and recycle assumptions. Without recycle of uranium or plutonium and with a 0.30% U-235 enrichment tails assay, about 260,000 MWe could be supported. Without recycle and at 0.20% tails assay, about 310,000 MWe could be supported. With recycle of uranium and plutonium and a 0.20% tails assay, about 520,000 MWe could be supported. All the levels of supportable capacity are above the 170,000 MWe of capacity in operation (55,000 MWe), under construction (95,000 MWe), and on order (20,000 MWe), as of late 1980. Thus, currently estimated resources can provide adequate uranium supplies for a sizable expansion to U.S. nuclear generating capacity.

The cumulative lifetime (30 years) uranium requirements for all of the above reactors (170,000 MWe) would be about .907 million tonnes (1.0 million tons) U_3O_8 at 0.20% enrichment tails with no recycle, compared to the 1.45 million tonnes (1.6 million tons) mean value in $13.60 (($30) or the 2.27 million tonnes at $22.65 (2.5 million tons at $50)) ore reserves, by-product, and probable potential resources. Evaluation of long-term fuel commitments on the basis of ore reserves and probable potential resources is considered a prudent course for planning. The lifetime commitment would be less than one-third of currently estimated $22.65 ($50) domestic resources, including the possible and speculative categories (see Table 9.1).

9.3.2.7 Uranium resource recovery

In regard to the availability of estimated uranium resources considering recoveries in mining and ore processing, estimates of U.S. uranium resources represent the quantity of uranium estimated to be minable expressed as tons of U_3O_8 of ore in the ground. These estimates are a reflection of the information available to DOE at the time of the estimate and thus are dependent on the extent of exploration. In view of the considerations involved in preparing the resource estimates and the uranium resource outlook, no adjustment for losses is warranted.

U.S. mining practice results in recovery of high percentages of the uranium contained in a deposit. DOE resource estimation procedures consider the capabilities and requirements of mining systems currently in use so that the estimates are a realistic appraisal of what is minable. Because deposits frequently are not fully delineated before they are developed, it is not unusual for more uranium to be recovered from deposits than was included in ore reserves before such deposits were put into production. Mining company practice seeks to recover as much of the contained mineral content as possible before abandoning a mine. A strong incentive for such practice is the increase in financial returns. In the processing of uranium ores, recoveries generally are over 90%. In 1980, mill recovery averaged about 93%. Higher recoveries are usually possible if economically justified.

9.3.2.8 High cost resources

An alternative to identification of additional low-cost resources is the utilization of higher cost resources. The highest cutoff cost category included in DOE resources in Table 9.1, is $45.30 per kilogram ($100 per pound) of U_3O_8. This level is an upper range of what might be of interest for utilization in light water reactors over the next few decades.

The increased price of oil and coal in the last few years has been a contributing factor to the increased price of uranium economically acceptable in light water reactors. This impact results from the relative insensitivity of nuclear electric power costs to increases in uranium prices. The cost of fuel is a very small fraction of the cost of power from a nuclear plant. In turn, the cost of natural uranium is only a small fraction of the fuel cost; enrichment, fabrication, reprocessing, and carrying charges make up the balance. As a result, large increases in uranium prices result in comparatively small increases in power costs. As pointed out in Section 9.3.2.6, nuclear capacity currently in operation, under construction, and on order, is expected to have adequate supplies of U_3O_8 at prices much lower than $45.30 per kilogram ($100 per pound) in 1980 dollars.

Knowledge of U.S. resources in the above $22.65 ($50) category is meager, largely because of the lack of past economic interest. There has been virtually no industry activity to search for or to develop such resources. Prospects for discovery of higher cost resources in the United States are considered promising at this stage of U.S. exploration. The principal large, very low-grade deposits that have been studied in some detail in the past are the shales and phosphates. The Chattanooga shale in Tennessee is of particular interest because of its large size. This deposit was extensively drilled, sampled, and studied in the 1950s. The higher grade part of the Chattanooga shale has an average uranium content of about 60 to 80 ppm compared to 1500 ppm in present-day ores. It contains in excess of 4.5 million tonnes (5 million tons) of U_3O_8 that may be producible at a cost of $45.30 or more per kilogram ($100 or more per pound) of U_3O_8. Additional work to develop production technology will be needed.

If Chattanooga shale were mined to fuel an 1150-MWe reactor, assuming recycle of uranium (but not of plutonium) and a 0.3% enrichment tail, about 11,428 tonnes (12,600 tons) of shale would have to be processed each day; with uranium and plutonium recycle (should that be practiced) and 0.20% enrichment tails, about 7,710 tonnes (8500 tons) per day would have to be processed. An average of about 10,250 tonnes (11,300 tons) of coal would have to be burned each day if 20 MJ/kg (8700 Btu/lb) of coal were used to produce power equivalent to that produced by a 1150-MWe reactor.

Utilization of the very low-grade resources such as Chattanooga shale would, of course, involve mining and processing very much larger quantities of ore than is currently mined to produce the same amount of uranium. From an environmental as well as from an economic point of view, identification and utilization of additional higher grade ores would be preferable. However, the shales are available if their use should become necessary.

9.3.2.9 Prices

During the period 1973-1979, the average delivery price per kilogram (pound) of U_3O_8 for sales from domestic producers to domestic buyers, in year-of-delivery dollars, increased from $3.22 to $10.80 ($7.10 to $23.85), as shown in Table 9.4.

Table 9.4. Historical trend of average uranium prices

(Dollars[a] per pound of U_3O_8)

Year	Final Price
1973	$ 7.10
1974	7.90
1975	10.50
1976	16.10
1977	19.75
1978	21.60
1979	23.85

[a] Year-of-delivery dollars.

(To convert dollars per pound to dollars per kilogram, multiply by 0.453.)

Future prices for material under contract as of July 1, 1980, as reported to DOE, is shown in Table 9.5. Also shown are the percentages of material under contract price arrangements covering the price data presented. The remainder is in market price contracts or in captive production.

9.3.2.10 Foreign uranium resource position

The most reliable source of information on world uranium resources is that compiled by the Working Party on Uranium Resources sponsored jointly by the Nuclear Energy Agency (NEA) and the International Atomic

Table 9.5. Average contract prices and settled market price contracts for uranium July 1, 1980

(Dollars[a] per pound of U_3O_8)

Year	Price	Percentages of procurement under contract price contracts
1980	26.00[b]	66
1981	28.70[b]	55
1982	34.80	47
1983	41.40	43
1984	43.45	35
1985	43.45	32
1986	46.85	16
1987	43.55	18
1988	42.70	22
1989	51.85	23
1990	53.25	16

[a] Year-of-delivery dollars.

[b] These years include settled market price contracts. Market price contract prices are determined sometime before delivery, based on prevailing market prices.

(To convert dollars per pound to dollars per kilogram, multiply by 0.453.)

Energy Agency (IAEA). This group has been gathering and publishing uranium resource estimates since 1965 and includes most of the significant uranium resource countries. In compiling its estimates this group classifies resources as Reasonably Assured resources (roughly comparable to ore reserves in the usual mining industry sense) and Estimated Additional resources (roughly comparable to DOE's probable potential resources). Resources in the world outside of the centrally planned economies area (WOCA) are tabulated by continents and major countries in Table 9.6.

Almost 80% of these resources are concentrated in three continents: North America, Africa, and Australia. Six countries within those continents - U.S., Canada, South Africa, Namibia, Niger, and Australia - have about three-quarters of the Reasonably Assured resources. This geographic concentration is a reflection of the geologic favorability of these areas as well as the extent of exploration and resource appraisal efforts to date.

9.3.2.11 Foreign production capacity and plans

Studies by the NEA and the IAEA have also provided reliable information on world production capacity. The current production capacity of existing non-U.S. plants (Class 1) is about 34,466 tonnes (38,000 tons) U_3O_8 annually, as shown in Table 9.7. This production is primarily in Canada, France, Namibia, Niger, and South Africa.

Construction of new plants (Class 2) with a capacity of about 7,256 additional tonnes (8,000 tons) is taking place, primarily in Australia and Canada. Plants that are planned (Class 3), could increase total annual production by another 32,652 tonnes (36,000 tons) U_3O_8 for a total of 76,188 tonnes (84,000 tons) U_3O_8 by 1990. Since needs for uranium are well below attainable production capacity levels, and prices would not justify all operations, it is likely that many of the projected plants will be built on a deferred schedule. It is also possible that some new plants will replace existing operations. Countries of particular significance in future production expansion are Australia and Canada, which have 82% of capacity under construction and 70% of the planned additional capacity.

9.3.2.12 Foreign reactor requirements

The uranium requirements in non-Communists foreign countries have been projected by the Energy Information Administration based on the reactors planned and timing of construction. Table 9.8 shows three cases of power plant growth which, by the year 2000, range from 300 GWe to 400 GWe of nuclear power in operation. The mid-case is taken as the most likely one. However, nuclear power growth projections have been subject to continual downward revision in the last several years.

Table 9.6. World uranium resources by continent[a]

(World outside centrally planned economies area)
(thousand tons U_3O_8)

	Reasonably assured		Estimated additional	
	$30/lb	$50/lb[b]	$30/lb	$50/lb[b]
	North America			
U.S.	645	940	885	1,430
Canada	280	305	480	945
Other	9	44	44	65
Total	930	1,290	1,410	2,440
	Africa			
South Africa	320	508	70	180
Niger	210	210	69	69
Namibia	152	173	39	69
Other	109	115	2	22
Total	790	1,000	180	340
	Australia			
Total	380	390	165	180
	Europe			
France	51	72	34	60
Spain	13	13	11	11
Sweden	1	390	0	4
Other	22	31	19	53
Total	90	510	60	130
	Asia			
India	39	39	1	31
Other	13	21	0	0
Total	50	60	0	30
	South America			
Brazil	96	96	117	117
Argentina	30	36	5	12
Other	0	0	7	8
Total	130	130	130	140
Worldwide total (rounded)	2,400	3,400	1,900	3,300

[a]Modified from "Uranium Resources, Production and Demand" OECD, Nuclear Energy Agency (NEA), and the International Atomic Energy Agency (IAEA), December 1979.
[b]Includes resources at $30 per pound of U_3O_8.

(To convert tons to tonnes, multiply by 0.907; to convert $ per pound to $ per kilogram, multiply by 0.453.)

In order tq supply these nuclear plants, EIA has estimated the amount of uranium required assuming 0.20% U-235 enrichment plant tails and no recycle of uranium or plutonium. Table 9.8 gives the annual tons U_3O_8 from 1980 to 2000 for high-, mid-, and low-cases.

For the mid-case, foreign requirements increase from 16,689 tonnes (18,400 tons) U_3O_8 in 1980, to 23,763 tonnes (26,200 tons) U_3O_8 in 1985, and to 49,069 tonnes (54,100 tons) U_3O_8 in the year 2000. Cumulative requirements through the year 2000 total 650,319 tonnes (717,000 tons) U_3O_8.

If all the planned foreign mine-mill production came on-stream as currently projected, there would be considerable excess capacity. If only operating mills or those under construction were available by the late 1980s, production capacity would cover annual demands through the late 1990s.

Table 9.7. Foreign uranium production capability

(Thousand tons U_3O_8)

Year	Australia			Canada			France			Namibia			Niger			S. Africa			Other[b]			Foreign Total		
	1[a]	2	3	1	2	3	1	2	3	1	2	3	1	2	3	1	2	3	1	2	3	1	2	3
1980	1.3	0	0	9.8	0	0	4.5	0	0	5.3	0	0	5.2	0	0	8.3	0	0	4.1	0	0	38.5	0	0
1981	1.8	1.1	0	9.8	1.4	0	4.5	0.2	0	5.3	0	0	5.2	0	0	8.3	0	1.2	4.1	0	0.8	39.0	2.7	2.0
1982	1.8	3.3	0	9.8	1.9	0	4.5	0.5	0	5.3	0	0	5.2	0	0	8.3	0	2.9	4.1	0	3.0	39.0	5.7	5.9
1983	1.8	3.3	0	10.5	1.9	2.0	4.5	0.7	0	5.3	0	1.2	5.2	0	0	8.3	0	4.6	4.1	0	4.1	39.7	5.9	11.9
1984	1.8	3.3	0	11.0	2.9	4.0	4.5	0.7	0	5.3	0	1.2	5.2	0	0.7	8.3	0	5.2	4.1	0	4.4	40.2	6.9	15.5
1985	1.8	3.3	6.5	12.0	2.9	5.0	4.5	0.7	0	5.3	0	1.2	5.2	0	2.5	8.3	0	5.5	4.1	0	5.1	41.2	6.9	25.8
1986	1.2	3.3	11.5	12.0	2.9	7.2	4.5	1.4	0	5.3	0	1.2	5.2	0	5.2	8.3	0	5.6	4.1	0	5.1	40.6	7.6	35.8
1987	1.2	3.3	11.5	12.0	2.9	7.2	4.5	1.4	0	5.3	0	1.2	5.2	0	5.2	8.3	0	5.6	4.1	0	5.2	40.6	7.6	35.9
1988	1.2	3.3	11.5	12.0	2.9	7.2	4.5	1.4	0	5.3	0	1.2	5.2	0	5.2	8.3	0	5.5	4.1	0	5.3	40.6	7.6	35.9
1989	1.2	3.3	11.5	12.0	2.9	7.2	4.5	1.4	0	5.3	0	1.2	5.2	0	5.2	8.3	0	5.5	4.1	0	5.4	40.6	7.6	36.0
1990	1.2	3.3	11.5	12.0	2.9	7.2	4.5	1.4	0	5.3	0	1.2	5.2	0	5.2	8.3	0	5.2	4.1	0	5.5	40.6	7.6	35.8
Total																							84.0	

[a]Class: 1. Currently operating plants
2. Plants under construction
3. Planned plants

[b]Includes: Argentina, Brazil, CAR, Gabon, India, Italy, Mexico, Portugal, Spain, Yugoslavia. Based on "Uranium Resources, Production and Demand," December 1979.

(To convert tons to tonnes, multiply by 0.907.)

Table 9.8. Foreign nuclear capacity and uranium requirements

	Capacity [GW(e)]			Requirements (tons U_3O_8)[a]		
	Low	Mid	High	Low	Mid	High
1980	66	68	77	17,300	18,400	19,800
1985	117	124	128	24,000	26,200	29,200
1990	165	181	201	27,500	31,600	32,700
1995	229	252	280	34,600	41,500	47,800
2000	300	350	400	42,700	54,100	64,300

[a]0.20% U-235 tails assay.

(To convert tons to tonnes, multiply by 0.907.)

Additional projections of WOCA nuclear growth and uranium requirements were developed during the International Nuclear Fuel Cycle Evaluation (INFCE). While the projections are now considered as high by many, they do provide an additional, more optimistic, viewpoint on future nuclear growth. The INFCE low case — modified to exclude the U.S. — indicated a growth in foreign (WOCA) nuclear capacity from 82 GWe at the end of 1980 to 217 GWe in 1990 and 580 GWe in the year 2000. Corresponding foreign uranium requirements would be 19,047 tonnes (21,000 tons) in 1980, 45,350 tonnes (50,000 tons) in 1990 and 108,840 tonnes (120,000 tons) in 2000. Such projections indicate a much larger possible growth in future uranium demands

9.3.2.13 <u>Foreign competition and the domestic industry</u>

The concentration of world uranium resources and production has, in past periods of low prices and ore production, fostered attempts to form cartel-like organizations seeking to restrict the free movement of uranium and influence pricing. The concentration of uranium production in a few countries will continue for some time, though there is an increasing diversity of supply sources. The opportunity for future foregin cartel-like activities will continue, particularly if uranium producer country governments are involved, which has been the case in the past. However, the severe criticism of such practice and the legal actions that have resulted in the United States might operate to discourage such activities in the future. Since the U.S. has the capability of producing a large portion — or all — of its uranium needs, and since the U.S. uranium buyers historically have shown a strong preference for domestic uranium, the U.S. is not expected to develop a large dependence on foreign uranium. These factors would tend to reduce the susceptibility of the U.S. to direct impacts of any cartel-like activity.

9.3.2.14 Conclusions

In conclusion, DOE assessment of uranium resources indicates that currently estimated ore reserves and probable potential resources at forward costs up to $13.60 per kilogram ($30 per pound) U_3O_8 total over 1.36 million tonnes (1.5 million tons), and at forward costs up to $22.65 per kilogram ($50 per pound) U_3O_8 total almost 2.17 million tonnes (2.4 million tons). The 2.17 million tonnes (2.4 million tons) U_3O_8 will support 390 GWe of nuclear power generating capacity, assuming a 30-year life for the reactors, no spent fuel reprocessing and an enrichment plant tails assay of 0.20% U-235. Under the latest DOE forecast for nuclear generating capacity in the post-2000 period, these resources should support U.S. nuclear power growth, including SONGS 2 and 3, well into the next century. However, meeting the uranium requirements for an expanding U.S. nuclear power industry will require extensive industry efforts to sustain exploration, and success in discovering and developing the potential uranium resources.

Foreign uranium resources are substantial and have been growing. Some of the more recently discovered deposits, especially in Canada and Australia, will have comparatively low-cost uranium production. The staff, therefore, concludes that there will be sufficient nuclear fuel available for SONGS 2 and 3.

9.4 DECOMMISSIONING

A license to operate a nuclear power plant is issued for a period of 40 years, beginning with the issuance of the construction permit. At the end of the 40-year period the operator of a nuclear power plant must renew the license for another time period or apply for termination of the license and for authority to dismantle the facility and dispose of its components.[8] If prior to the expiration of the operating license, technical, economic, or other factors are unfavorable to continued operation of the plant, the operator may elect to apply for license termination and dismantle authority at that time. In addition, at the time of applying for a license to operate a nuclear power plant, the applicant must show that he possesses "or has reasonable assurance of obtaining the funds necessary to cover the estimated costs of permanently shutting the facility down and maintaining it in a safe condition."[9] These activities, termination of operation and plant dismantling, are generally referred to as "decommissioning."

NRC regulations do not require the applicant to submit decommissioning plans at the time the construction permit and operating license are obtained; consequently, no definite plan for the decommissioning of the station has been developed. At the end of the station's useful lifetime, the applicant will prepare a proposed decommissioning plan for review by the Nuclear Regulatory Commission. The plan will comply with NRC rules and regulations then in effect.

The decommissioning of reactors is not new. Since 1960, five licensed nuclear plants, four demonstration nuclear power plants, six licensed test reactors, 28 licensed research, and 22 licensed critical facilities have been or are in the process of being decommissioned.[10] The primary methods of decommissioning consist of mothballing, entombing, dismantling, or a combination of these three alternatives. The primary methods are defined below in terms of the definitions provided in Regulatory Guide 1.86.[11]

Mothballing is the process of placing a facility in a nonoperating status. The reactor may be left intact except that all reactor fuel, radioactive fluids, and nonfixed radioactive wastes such as ion exchange resins, contaminated scrap materials, and contaminated chemicals are removed. The existing license is amended to a "possession only" status and continues in effect until residual radioactivity decays to levels acceptable for release to unrestricted access or until residual radioactivity is removed. The "possession only" license is a reactor facility license that permits a licensee to possess the facility but prohibits operation of the facility as a nuclear reactor.

Entombment consists of removing all fuel assemblies, radioactive fluids, and wastes followed by the sealing of remaining radioactive material within a structure integral with the biological shield or by some other method to prevent unauthorized access into radiation areas. A program of inspection, facility radiation surveys, and environmental sampling is required for a licensed facility that has been entombed.

Dismantling is defined as removal of all fuel, radioactive fluids and waste, and all radioactive structures. Surface contamination levels, established in Table 1 of Regulatory Guide 1.86, must be met prior to termination of the facility license. In addition to meeting the surface contamination levels, the acceptability of the presence of materials which have been made radioactive by neutron activation would be evaluated on a case-by-case basis prior to termination of the license. If the facility owner so desires, the remainder of the reactor facility may be dismantled and all vestiges removed and disposed of.

For a single nuclear reactor, the mothballing alternative costs about $2.45 million initially plus an annual maintenance and surveillance cost of $167,000. If a 24-hr manned security force is not required (e.g., a site with continuing operations), the annual cost could be reduced to $88,000. Translating these costs into unit cost of generating electricity, the 30-year levelized unit cost* would be about 0.04 mills/kWhr and if a manned security force is not required about 0.03 mills/kWhr.[12]

The entombing alternative costs about $7.58 million initially for a single unit facility plus an annual maintenance and surveillance cost of $58,000 for the duration of the entombment period.[13] These costs, when translated to a 30-year levelized unit cost* basis, amount to about 0.06 mills/kWhr.

The dismantling alternative for a single nuclear power reactor costs about $33.3 million to remove the radioactive structures associated with NRC requirements for terminating a possession only license.[12] An additional $4.8 million would be needed to remove the nonradioactive structures (cooling towers, administration buildings, etc.) to below grade.[13] There are no annual costs associated with this alternative. When the dismantling costs are translated to a 30-year levelized unit cost* basis, this amounts to about 0.17 mills/kWhr.

Combinations of mothballing and delayed (about 100 years) dismantling have 30-year levelized unit costs that are about the same as the mothballing alternative costs. Likewise, the costs for the entombing delayed dismantling combinations are about the same as the entombing cost. In both instances the annual maintenance cost for mothballing and entombing alternatives, on a present value basis, is sufficient to cover all the delayed dismantling cost for the mothballing alternative and about 80% for the entombing alternative.

Although the above costs are for a one-unit station, the savings associated with multiunit stations are small; thus, the unit cost (mills/kWhr) is essentially the same for a single unit station or multiunit station. For the San Onofre Nuclear Generating Station Units 2 and 3 the decommissioning costs would be about double that indicated for all of the decommissioning one-unit alternatives.

Studies of social and environmental effects of decommissioning large commercial power generating units have not identified any significant impacts.[13]

Also, studies indicate that occupational radiation doses can be controlled to levels comparable to occupational doses experienced with operating reactors through the use of appropriate work procedures, shielding, and remotely controlled equipment.[13]

The applicant may retain the site for power generation purposes indefinitely after the useful life of the station. The degree of dismantlement would be determined by an economic and environmental study involving the value of the land and crop value versus the complete demolition and removal of the complex. In any event, the operation will be controlled by rules and regulations in effect at the time to protect the health and safety of the public.

SONGS 2 and 3 are designed to operate for about 30 years, and the end of their useful life will occur approximately in the year 2011. The applicant has made no firm plans for decommissioning, but assumes that the following steps would be taken as minimum precautions for maintaining a safe condition:

1. All fuel would be removed from the facility and shipped offsite for disposition.

2. All radioactive wastes – solid, liquid, and gas – would be packaged and removed from the site insofar as practical.

A decision as to whether the station would be further dismantled would require an economic study involving the value of the land and scrap value versus the cost of complete demolition and removal of the complex. However, no additional work would be done unless it is in accordance with rules and regulations in effect at the time.

In addition to personnel required to guard and secure the station, concrete and steel would be used to prevent ingress into any building, particularly the radioactive areas.

Since the San Onofre site is located on U.S. Marine Corps property, the applicant must, if desired by the government, remove all of the improvements installed on the site at the end of the applicant's lease arrangement. This requirement could potentially entail complete removal and dismantling of the plant (ER, Section 5.8).

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*Based on a 1200-MWe generating unit beginning operation in 1958, a capacity factor of 60%, an escalation rate of 5%, and a discount rate of 10%. A more complete analysis of decommissioning costs can be found in Appendix B of NUREG-0480.[6]

REFERENCES

Unless otherwise noted, documents are available in public technical libraries.

1. U.S. Department of the Interior, Bureau of Mines, "Mineral Facts and Problems," 1970, p. 230.

2. U.S. Atomic Energy Commission, "Uranium Industry Seminar," Grand Junction, Colorado, Office, Report GJO-108(74), October 1974.***

3. Energy Research and Development Administration, "Survey of U.S. Uranium Marketing Activity," Report ERDA 77-46, May 1977.***

4. U.S. Atomic Energy Commission, "Survey of U.S. Uranium Marketing Activity," Report WASH-1196(74), April 1974.***

5. U.S. Nuclear Regulatory Commission, "Final Generic Environmental Statement on the Use of Recycle Plutonium in Mixed Oxide Fuel in Light Water Cooled Reactors," Report NUREG-0002, vol. 4, U.S. Government Printing Office, Washington, D.C., August, 1976, Table XI-32, Section 4.**

6. U.S. Nuclear Regulatory Commission "Coal Vs. Nuclear: A Comparison of the Cost of Generating Baseload Electricity by Region," NUREG-0480, December 1978.**

7. U.S. Atomic Energy Commission, Press Release, No. T-517, Oct. 25, 1974.*

8. Title 10, "Rules and Regulations," Code of Federal Regulations, Part 50, "Licensing of Production and Utilization Facilities," Section 50.51, "Applications for Termination of Licenses."

9. Title 10, "Rules and Regulations," Code of Federal Regulations, Part 50, "Licensing of Production and Utilization Facilities, Section 50.33, "Content of Applications; General Information."

10. P. B. Erickson and G. Lear, "Decommissioning and Decontamination of Licensed Reactor Facilities and Demonstration Nuclear Power Plants," presented at Conference on Decontamination and Decommissioning in Idaho Falls, Idaho, Aug. 19-21, 1975.*

11. U.S. Nuclear Regulatory Commission, Regulatory Guide 1.86, "Termination of Operating Licenses for Nuclear Reactors."

12. U.S. Nuclear Regulatory Commission, "Draft Generic Environmental Statement on Decommissioning of Nuclear Facilities," USNRC Report NUREG-0586, January 1981.

13. Atomic Industrial Forum, Inc., "An Engineering Evaluation of Nuclear Power Reactor Decommissioning Alternatives," Report AIF/NESP-009.

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* Available for inspection and copying for a fee in the NRC Public Document Room, 1717 H Street, N.W. Washington, D.C. 20555.

** Available from NRC/GPO Sales Program, Washington, D.C. 20555, and the National Technical Information Service, Springfield, VA 22161.

*** Available from NTIS only.

10. BENEFIT-COST SUMMARY

10.1 RÉSUMÉ

There have been minor changes in the benefit-cost analysis of station operation since the issuance of the FES-CP in March 1973. The most significant changes are that the staff has revised the economic cost estimates upwards to reflect the rapid escalation seen during the last few years and has included among the benefits of station operation the fuel oil savings that will be made possible by having additional non-oil-fired, base-load capacity available in the California Power Pool. There have been essentially no significant changes in the staff's assessment of the environmental costs of operating SONGS 2 & 3; however, a broadening of the review process has occurred and is reflected in Table 10.1.

10.2 BENEFITS

The primary benefits of station operation will be the 9.3 to 13.0 billion kWhr of electricity produced by the two additional units each year (assuming a range of capacity factors of 50 to 70%), the increase in the reliability of electric service resulting from the addition of 2114 MWe of generating capacity, and an estimated regional decrease in the consumption of fuel oil of 13.2 million to 18.5 million barrels of oil per year (again assuming a range of capacity factors of 50 to 70%).

The staff also notes that operation of SONGS 2 & 3 will result in the generation of local revenues through property taxes and state sales and use taxes (annual property taxes will be approximately $13 million while state sales and use taxes resulting from station operation are estimated to be $3 million per year) and will increase local employment (over 300 new jobs will be directly created, with the average income of station workers being approximately $30,000 per year in 1980 dollars). However, these considerations are not included in the staff's benefit-cost analysis because from a societal viewpoint these local benefits are in actuality transfer payments from those using the electricity generated by the station to the recipients of the tax and employment benefits.

10.3 ECONOMIC COSTS

Since the issuance of the FES-CP the fuel, operating, and maintenance costs of nuclear plant operation have escalated more rapidly than anticipated by the staff in 1973. Based on more recent information, the staff now estimates the 1983 costs of station operation to be as follows: fuel costs — $120 million per year; operating and maintenance costs — $45 million per year; and decommissioning costs — $2.7 million per year.

10.4 ENVIRONMENTAL COSTS

Since the issuance of the FES-CP the applicants have accumulated additional environmental data and have made modifications in the station design. The staff, on making a reassessment of the environmental costs of station operation based on this new information, has found that the conclusions reached in the FES-CP are still valid. Table 10.1 summarizes the staff's assessment of the environmental impacts of station operation.

10.5 SOCIAL COSTS

The restriction in public use of 1.4 km (0.85 mile) of the San Onofre Beach is a significant cost of operation of the station. The number of personnel needed to operate SONGS 2 & 3 is a small fraction of the expected population growth in the communities near the station. As a result, the extra cost of providing public services to station personnel who relocate in the area is not likely to be discernible in these communities.

Table 10.1. Benefit-cost summary for the operation of SONGS 2 & 3[a]

Primary impact and population or resource affected	Unit of measure	Magnitude of impact
DIRECT BENEFITS		
ENERGY (50–70% capacity factor)	kWhr/yr $\times 10^6$	9,300–13,000
CAPACITY	kW $\times 10^3$	2,114
REDUCED FUEL OIL CONSUMPTION (50–70% capacity factor)	bbl/yr $\times 10^6$	13.2–18.5
ECONOMIC COSTS		
OPERATING (1980 dollars, 60% capacity factor)		
Fuel	\$/year	120,000,000
Operation and maintenance	\$/year	45,000,000
DECOMMISSIONING (annualized value)	\$/year	1,100,000
ENVIRONMENTAL COSTS		
IMPACT ON LAND		
Land use		Insignificant
Terrestrial ecology		Negligible
IMPACT ON WATER		
Fresh water consumption	gal/day	1,066,000
Salt water consumption		Insignificant
Heat discharge to natural water body		
Aquatic biota		Insignificant
Migratory fish		Insignificant
Chemical discharge to natural water body		
People		Not discernible
Aquatic biota		Not discernible
Water quality		Not discernible
Radionuclide contamination of natural surface water body		
All except tritium	Ci/year/unit	1.1
Tritium	Ci/year/unit	300
Chemical contamination of groundwater		
People		Not discernible
Plants		Not discernible
Radionuclide contamination of groundwater		
People		Not discernible
Plants and animals		Not discernible
Effects on natural water body of condenser cooling system operation		
Primary producers and consumers		Small
Fisheries		Small
Natural water drainage		
Flood control		No damage
Erosion control		Insignificant
IMPACT ON AIR		
Chemical discharge to ambient air		
Air quality, chemical		Negligible
Air quality, odor		Negligible
Radionuclides discharged to ambient air		
Noble gases	Ci/year/unit	8,800
Radioiodines	Ci/year/unit	0.195
Carbon-14	Ci/year/unit	8
Argon-41	Ci/year/unit	25
Tritium	Ci/year/unit	1,100
Particulates	Ci/year/unit	0.34
Fogging and icing		None
TOTAL BODY DOSES TO U.S. POPULATION		
General public, unrestricted area	Man-rem/year	442
SOCIETAL COSTS		
OPERATIONAL FUEL DISPOSITION		
Fuel transport (new)	Trucks per year	11
Fuel storage		In-building storage
Waste products (spent fuel)	Trucks per year	200
PLANT LABOR FORCE		Insignificant

[a]See Appendix C for calculations and explanations of table entries.

(To convert gal to liters, multiply by 3.7.)

10.6 ENVIRONMENTAL COSTS OF THE URANIUM FUEL CYCLE AND TRANSPORTATION

The staff has evaluated the environmental impacts of the uranium fuel cycle as presented in Table 5.8 and has found these impacts to be sufficiently small so that when superimposed upon the other environmental impacts assessed against the operation of the station, they do not alter the overall benefit-cost balance.

10.7 SUMMARY OF BENEFIT-COST

As the result of this second review of potential environmental, economic, and social impacts, the staff has been able to forecast more accurately the effects of station operation. The higher economic costs identified by the staff would not alter the staff's previous conclusion as to the overall balancing of the benefits of the station versus the environmental costs, whereas the benefit from the reduction in the regional consumption of fuel oil is felt to add significantly to the total benefits of station operation. Additional environmental costs have been identified as: (1) removal of approximately 1.4 km (0.85 mile) of beach from unrestricted public use, (2) possible destruction of at least a portion of the San Onofre Kelp Bed during the summer months by the heated water discharge, (3) occupation of about 7.2 ha (17.8 acres) of land by new towers, access roads, and switchyards associated with new transmission facilities, (4) environmental effects of the uranium fuel cycle as enumerated in Table 5.8, and (5) environmental impacts of transportation of fuel and waste to and from nuclear power plants as indicated in Table 5.7. Consideration of these additional costs together with those identified in the FES-CP does not alter the position taken in the FES-CP that the environmental and social costs are acceptable and that these costs are outweighed by the primary benefits of operating SONGS 2 & 3.

11. DISCUSSION OF COMMENTS RECEIVED ON THE DRAFT ENVIRONMENTAL STATEMENT

Pursuant to 10 CFR Part 51, the Draft Environmental Statment and a supplement to the Draft Environmental Statement related to the operation of the San Onofre Nuclear Generating Station, Units 1 and 2, were transmitted, with a request for comments, to:

Department of Agriculture

Department of Army (Corps of Engineers)

Department of Commerce

Department of Energy

Department of the Interior

Department of Health, Education, and Welfare

Department of Housing and Urban Development

Department of Transportation

Environmental Protection Agency

Federal Energy Regulatory Commission

Advisory Council on Historic Preservation

California Department of Health (Water Pollution Control Commission, Air Pollution Control Commission, Occupational Health Office)

California Department of Natural Resources

California Department of Parks and Recreation

In addition, the NRC requested comments on the Draft Environmental Statement and its supplement from interested persons by a notice published in the Federal Register on December 6, 1978 (43 FR 25183) and January 13, 1981 (46 FR 7435), respectively. In response to the request referred to above, comments were received from:

Draft Environmental Statement

U.S. Department of Agriculture, Science and Education Administration (DASEA)
U.S. Department of Agriculture, Economics, Statistics and Cooperative Services (DAESC)
U.S. Department of Agriculture, Soil Conservation Service (DASCS)
Department of Housing and Urban Development (HUD)
Department of the Army, Corps of Engineers (COE)
Federal Energy Regulatory Commission (FERC)
U.S. Department of the Interior (DOI)
Rourke and Woodruff Law Offices (RWL)
U.S. Department of Commerce (DOC)
Department of Health, Education, and Welfare (HEW)
Southern California Edison Company (SCE)
U.S. Environmental Protection Agency (EPA)
Mr. Marvin I. Lewis (MIL)
Richard J. Wharton (RJW)

Supplement to the Draft Environmental Statement

U.S. Dearptment of Agriculture, Economics, Statistics and Cooperative Services (DAESC)
Federal Energy Regulatory Commission (FERC)
U.S. Department of the Interior (DOI)
U.S. Environmental Protection Agency (EPA)
Union of Concerned Scientists (UCS)
Richard J. Wharton (RJW)
Southern California Edison Company (SCE)
Frank H. Grundel (FHG)
San Diego Association of Governments (SAG)
U.S. Department of Agriculture, Soil Conservation Service (DASCS)

The comments are reproduced in this statement as Appendix A. The staff's consideration of the comments received and its disposition of the issues involved are reflected in part by changes in the text in the pertinent sections of this Final Environmental Statement and in part by the discussion in Section 11. The comments are categorized by subject and are referenced by the use of the abbreviations indicated above. The pages in Appendix A on which copies of the respective comments appear are indicated by each subject title relating to the comment, and in the index to Appendix A.

11.1 EROSION CONTROL (DASCS, A-4)

The applicant's erosion control plans are briefly discussed and referenced in Sections 6.2.2 and 6.3.2. In addition, the treatment of disturbed areas is addressed in the FES-CP. Such discussions are beyond the scope of the OL review.

11.2 LOSS OF PRIME LANDS (DASCS, A-4)

The discussion of prime lands lost to access roads and transmission towers, which is presented in Appendix E of the DES, is based on available information as a result of staff discussions with Mr. Jack Smith and Mr. Ted Thee of the Soil Conservation Service (SCS) Escondido Field Station and Mr. Jon Christianson of the SCS Tustin Field Office.

11.3 RECREATION RESOURCES (DOI, A-5; RJW, A-48)

The original plan to allow recreational use in the beach area immediately adjacent to the nuclear plant was altered in the course of hearings before the ASLB and ASLAB based on safety considerations. The staff reasserts its judgment that, while significant, the impact of beach closure is not sufficient to warrant denying the applicant an Operating License for SONGS 2 & 3. While the 1.4 km (0.85 mile) of beach to be closed must be considered a valuable recreational resource, there are approximately 5.6 km (3.5 miles) of State Beach immediately south of this area and almost 1.1 km (0.7 miles) immediately north which remain open to the public. Of those three parcels of beach, the one to be closed gets substantially less use than the other two and is directly adjacent to the SONGS complex, where the natural aesthetics of the area have been altered by plant development. Finally, while the 30 years of beach closure is clearly a long time, it does not represent an "irreversible and irretrievable commitment of resources" as the intervenors contend. For these reasons, it is the judgment of the staff that the closure of this stretch of beach, while significant, is not sufficiently adverse to warrant forbidding plant operations.

11.4 RADIOLOGICAL IMPACTS (HEW, A-10; SCS, A-30; EPA, A-40/A-42; MIL, A-45; RJW, A-50)

The NRC staff agrees it is appropriate to note that dose commitments to any individual will also meet EPA regulation 40 CFR 190 which requires that such doses will not exceed 25 mrem/year to any individual.

The recent AIF study* referred to in this comment was an effort to provide the potential impact of lowering the exposure limit to 500 millirems per year. The data were developed to fit the model that the AIF developed to evaluate the impact of the exposure limit reduction. The exposure data were meant to portray the type of exposure situations which occur at PWR's but are not likely to occur every year at each plant. (See Section 5.5.1.4 for staff consideration of occupational radiation exposures.)

Table 5.8 is based on NRC Table S-3, from 10 CFR Part 51, and is a generic discussion of impacts for the balance of the uranium fuel cycle. The staff is bound by the Commission standard as shown in Table 5.8. A discussion of alternative handling of HLW or TRU wastes is therefore inappropriate for considerations of licensing SONGS 1 & 2.

*"A Preliminary Assessment of the Potential Impacts on Operating Nuclear Power Plants at a 500 millirem per Year Occupational Exposure Limit," J. Vance, C. Weaver, E. Lepper, AIF, April 1978 (unpublished).

The staff has made its own independent and reasonably conservative estimates of potential doses to maximum individuals as a result of the operation of SONGS 2 & 3. Considering the uncertainties involved in such calculations, the staff finds a factor of 3 difference to be in very good agreement. Therefore, the staff rejects the request that Table 5.4 of the DES be revised in order to be consistent with applicant's estimated doses.

The staff calculation was for sport fish taken in the mixing zone, not 0-10 miles from the outfall, and is an independent and a reasonably conservative estimate of doses to a maximum individual. It is true that doses would be much less at greater distances from the outfall. However, the staff rejects the suggestion that Tables 5.6 (and 5.2 and 5.3) are in need of revision in order to be consistent with the applicant's estimates, particularly when both sets of estimates are orders of magnitude below the Appendix I design objective doses.

The staff agrees that the DES contains relatively little information regarding beach use at the SONGS site. Detailed discussion is presented in the Applicant's ER (e.g., pp. 2.1-4 to 2.1-7, and 5.2-1 through 5.2-54). In addition, more information regarding the staff conclusions and assumptions relating to doses to transient populations at the beach is presented in response to EPA comments.

The dose to individual users of the beach was not calculated for the following reasons:

1. The prevailing wind direction generally carries radioactive effluents away from the beach, thereby lowering potential exposures.

2. The beach is part of the exclusion area of the plant site, and public use (e.g., sunbathing and picnicking) is not permitted (e.g., see p. 2.105 of the applicant's ER).

3. The walkway connecting the south and north beaches is at the bottom of a 28-ft seawall which effectively shields passerbys from any direct radiation from the plant.

4. Although the dose rates at the site boundaries are expected to be low, annual doses to individuals would be even lower due to limited exposure times in transit between beaches.

Doses to individuals at the visitor center (0.1 mi E) were calculated, but occupancy factors result in much lower annual doses than calculated from permanent residents assumed living year-around at the WNW site boundary (0.36 mi) reported in the DES. As noted in Section 5.5.1.4, direct radiation (other than from the gaseous plume) from SONGS 2 & 3 is expected to be very low at the beach area. When coupled with limited exposure periods for transients, and shielding from the 28-ft-high seawall, the potential annual doses would be insignificant.

Population doses included transient populations by sector within 10 miles of the site. Transient populations were added to the projected resident populations for the year 2000 by assuming each transient spent one full day (24 hours) visiting during each year.

10 CFR Part 20 (10 CFR 20.105a) has been modified to include the provisions of 40 CFR Part 190. Also, the SONGS 2 & 3 technical specifications will require a demonstration to show compliance with 40 CFR Part 190 considering the operation of three reactors at the SONGS site.

Section 5.5.3 of the FES has been modified to include the long-term environmental effects associated with carbon-14, krypton-85, and tritium releases of the fuel cycle excluding the reactor releases. These modifications were added to the earlier discussion which focused largely on the radon-222 impacts.

Staff estimates of the longer term effects of carbon-14, tritium, krypton-38, and releases of the reactor contribute less than 30% of the total fuel cycle impacts presented in Section 5.5.3 of the FES. Health effects reported in the FES on a "per reactor year" basis can be multiplied by the reactor operating time (i.e., 30 years) to obtain the total or integrated estimate.

Nevertheless, the staff is in the process of modifying its calculation methodology to automatically consider the radiological impacts of effluent releases of the entire nuclear fuel cycle.

It is important to note that the FES results conservatively include the impacts of both uranium and plutonium recycle even though such operations are not currently permitted. Thus, the FES results are conservative for any recycle option, especially the "throw-away" cycle, the option currently allowed.

The NRC staff has reevaluated the proposed preoperational radiological environmental monitoring program for SONGS 2 and 3. The proposed program is based on the existing SONGS 1 operational program. That program will be revised in the near future to meet the Appendix I (10 CFR Part 50) requirements now being incorporated into the Environmental Technical Specifications for Unit 1, thereby updating the preoperational program for Units 2 and 3.

Response to specific EPA comments are as follows:

1. Current NRC criteria require collection and measurement of I-131 only, since it is the radioiodine which accounts for essentially all of the radioiodine environmental dose commitment (nearly all of which is through food pathways). The reason the applicant specified a maximum of 8 days was to be certain that the samples can be collected, transported to a laboratory (often at a considerable distance), and analyzed "within 8 days" under difficult circumstances (e.g., storms, trucking strikes, etc.). In most cases the elapsed time will be much less.

2. The intent of the air sampling program is to monitor continuously at all sites. However, experience has shown that occasionally air sampling equipment fails during a 7-day period, and the samples are of no value. The same experience indicates the applicant can almost always achieve 75-80% reliability.

   That is the only reason for mention of "a minimum of 10 samples per quarter" by the applicant.

3. The staff agrees that it might be desirable to have a TLD station along the walkway below the seawall. However, as noted in response to the previous comment, the walkway is 28 ft below the top of the seawall and there is no line-of-sight between the beach and any radiation sources on the site. The beach in front of the site is part of the exclusion area (i.e., no sunbathing, picnicking, etc. is permitted). Therefore, there is no possibility of any member of the public receiving a measurable radiation dose since individual exposure times would be very small.

4. The NRC has included U.S. population dose commitment estimates in Section 5.5 for a few years (see, for example, Table 5.5). In addition, the staff has been including discussion of the Rn-222 question since mid-1978 (see Section 5.5.3).

Table 5-8 says Rn-222 releases are "Presently under reconsideration by the Commission," and in footnote a, "These issues which are not addressed at by this table may be subject to litigation in individual licensing proceedings." The results of generic testimony by the staff at other hearings is summarized briefly on pp. 5-36 to 5-40. Contrary to Mr. Lewis' assertion, NEPA does not require quantification of the impacts of Rn-222 releases over the "full period of toxicity" (presumably he is referring to Th-230, the parent of Rn-222). The staff feels the conservative evaluation in Section 5.5.3 probably accounts for the releases and potential doses resulting from Rn-222 releases over periods of many millenia. In addition, the FES will provide a revised Section 5.5.3 which also includes potential impacts of C-14 over periods up to 1,000 years into the future.

The staff agrees that stabilization of surface tailings piles cannot be assured "forever." However, numerous options, including deep burial in worked-out open pit mines or underground mines, are being voluntarily used by some applicants and may be used increasingly in the future by others. It should be noted that if such tailings are exposed by acts of God (e.g., glaciation), the potential long-term impacts could be lower than for the natural uranium ore since milling will remove over 90% of the U-235 (the parent of Th-230).

Environmental releases from "nuclear waste materials, including the interim storage of spent fuel, on site" are so small relative to normal operational releases as to be inconsequential. Such releases and potential impacts have been estimated* and do not significantly change the estimated impacts presented in the DES for normal operations. In that sense, the releases have been included in the DES assessment of environmental impacts.

11.5 METEOROLOGY (SCE, A-16)

The onshore tracer test results indicated that measured ground-level centerline one-hour average concentrations were less than concentrations estimated from the usual staff calculations. But a reduction for annual average values would not be the same.

*Draft Generic Environmental Impact Statement on "Handling and Storage of Spent Light Water Power Reactor Fuel," NUREG-0404, U.S. Nuclear Regulatory Commission (March 1978).

Over a long period of time, such as a year, the wind direction within a sector should be randomly distributed, and thus the path of the plume should be randomly distributed. The time that any part of the plume is over a point within the sector contributes to the annual average at that point. This time integration and random path of plume have the net effect of uniformly distributing an effluent horizontally within a sector. Any enhanced dispersion due to additional horizontal plume spreading would not reduce the annual average concentration. In the staff calculations it was assumed that the effluent was uniformly distributed horizontally over the sector.

The ground-level annual average concentration would be dependent on the wind frequency and on the vertical distribution of the effluent within the plume. No direct measurements of vertical plume distributions were made during these onshore tests or the tests referenced in NUS-1927. Without a more definitive description of the vertical distribution of the plume, it has been the staff position not to adjust annual average dispersion estimates.

11.6 THERMAL ANALYSIS (SCE, A-20/A-21)

The air temperatures used by the staff in its thermal model are too high and, therefore, the staff's ambient temperature predictions are too high. However, nonlinear effects of the air temperature on the water temperature are negligible so that the staff's excess temperature predictions are correct despite the systematically high ambient air temperatures used in the mathematical model.

Higher near-shore predicted ambient water temperatures appear as a result of the depth-averaged format of the predictions. The near-shore region is shallow, resulting in near homogeneous vertical temperature structure. In deeper water, strong stratification is present so that the depth-averaged water temperatures are lower due to the presence of cool bottom water. The staff's actual ambient surface temperature predictions show no variation in the offshore direction. Ambient temperatures based on field measurement have been used in Section 5.4 of the FES.

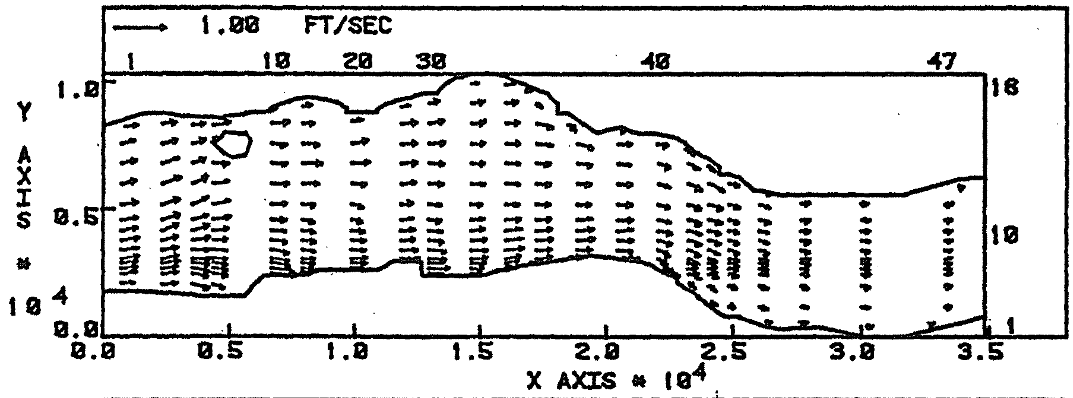
A brief discussion of the applicant's error analysis is given in Section 5.3.1.1 of the FES.

Errors in the staff's mathematical model can be introduced in several ways. First consider the accuracy of the numerical method. The TEMPTWO algorithm is consistent and stable. The use of direct upwind differencing can produce numerical dispersion when the Courant Number is not equal to 1. To minimize this error, the staff used a time-step that produced a Courant No. of 1 near the diffusers. This essentially eliminated numerical dispersion error around the discharge areas. Far from the discharges, numerical dispersion exists; however, this makes the predictions less conservative. To correct for this, the staff could raise the predicted excess temperatures in the far field. The staff believes that inaccuracies due to numerical dispersion are slight and, therefore, corrections of such inaccuracies are unnecessary.

Errors could also have been introduced through the methods used to represent turbulent transport and surface heat transfer. In developing the model, an effort was made to incorporate formulations which are universal; that is, to create a model that requires no adjustment of coefficients on a site-specific basis. The TEMPTWO model has been applied to other plants and results have compared favorably with available data. Thermal predictions for the Peach Bottom Plant and the Anclote Plant, including comparison with field data, are shown in Figures 11.1 through 11.8. The Peach Bottom Plant is on an impoundment of a river in Pennsylvania and the Anclote Plant is located on the Gulf of Mexico. The success of the TEMPTWO model at two quite different sites indicates that the submodels for turbulent transport and surface heat transfer can confidently be applied to San Onofre.

The staff's model did not include plant-induced densimetric flows. The staff performed a scale analysis and determined this effect to be insignificant at the San Onofre site.

Individual jet mixing is not calculated within the staff's model. However, this effect was included based on the applicant's near-field results.



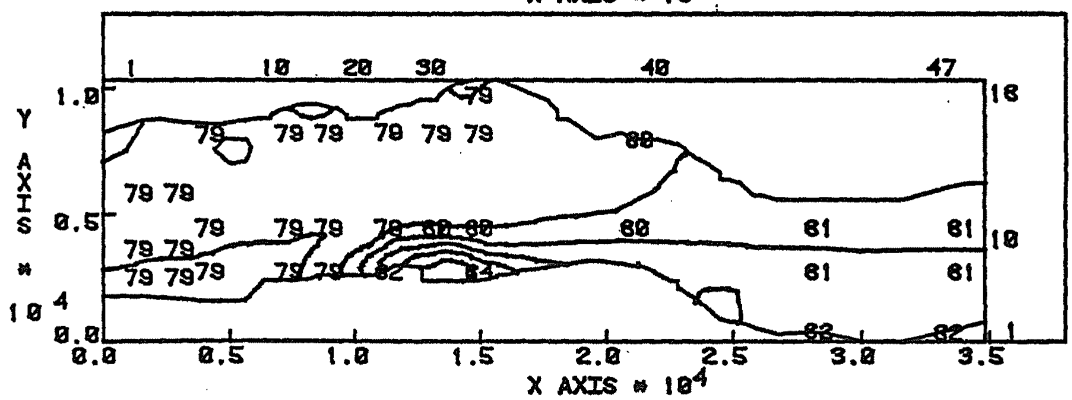


Figure 11.1 Computer simulation results for the two-dimensional depth-averaged (with self-similar vertical variation) flow conditions and temperature conditions (isotherms with 1°F gradation (1/1.8°C) in the Conowingo Pond Reservoir in the vicinity of the Peach Bottom Atomic Power Station at 9 a.m. on July 18, 1974, during reservoir conditions: downstream low flow after slack water.

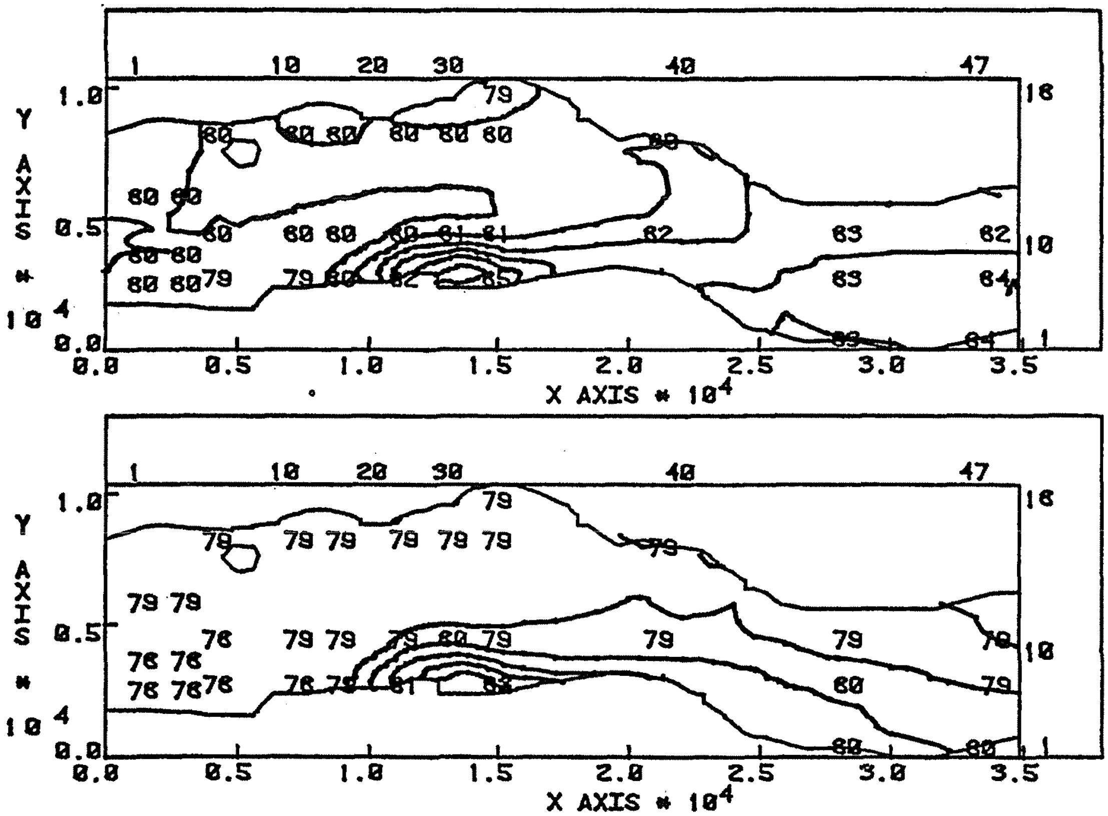


Figure 11.2 Computer simulation results for the surface and bottom temperature conditions (isotherms with 1°F gradation (1/1.8°C) in the Conowingo Pond Reservoir in the vicinity of the Peach Bottom Atomic Power Station at 9 a.m. on July 18, 1974, during reservoir conditions: downstream low flow slack water.

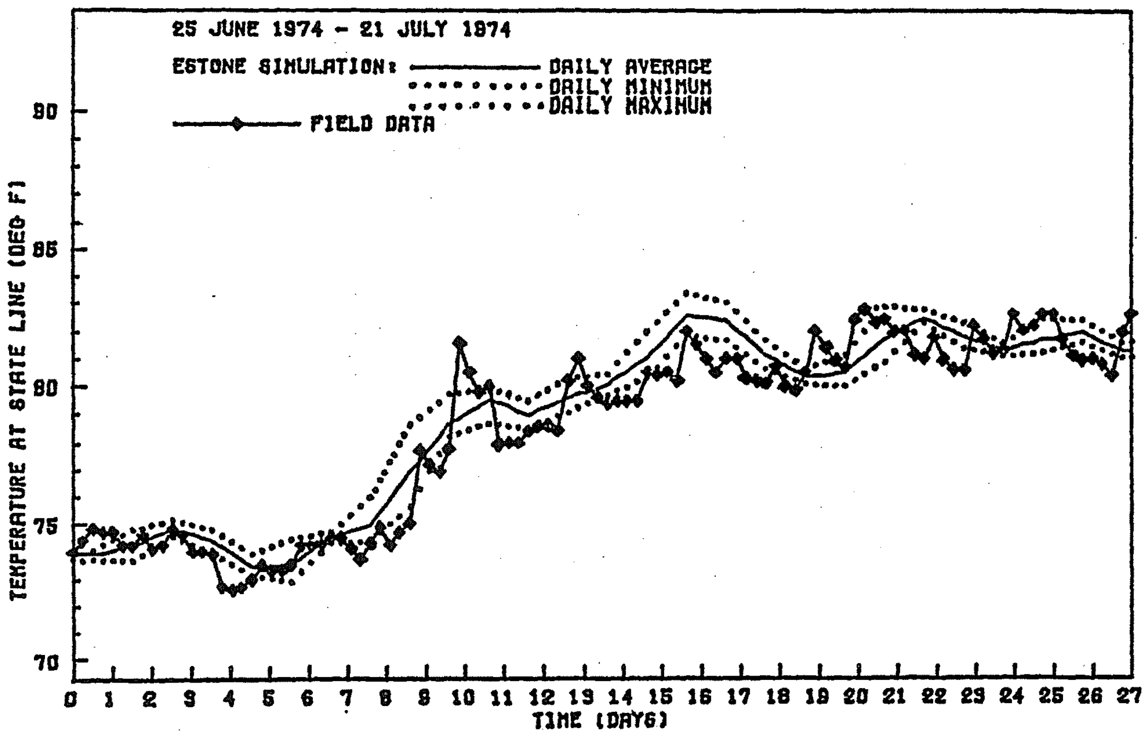


Figure 11.3 Estone simulation June 25, 1974 through July 21, 1974.

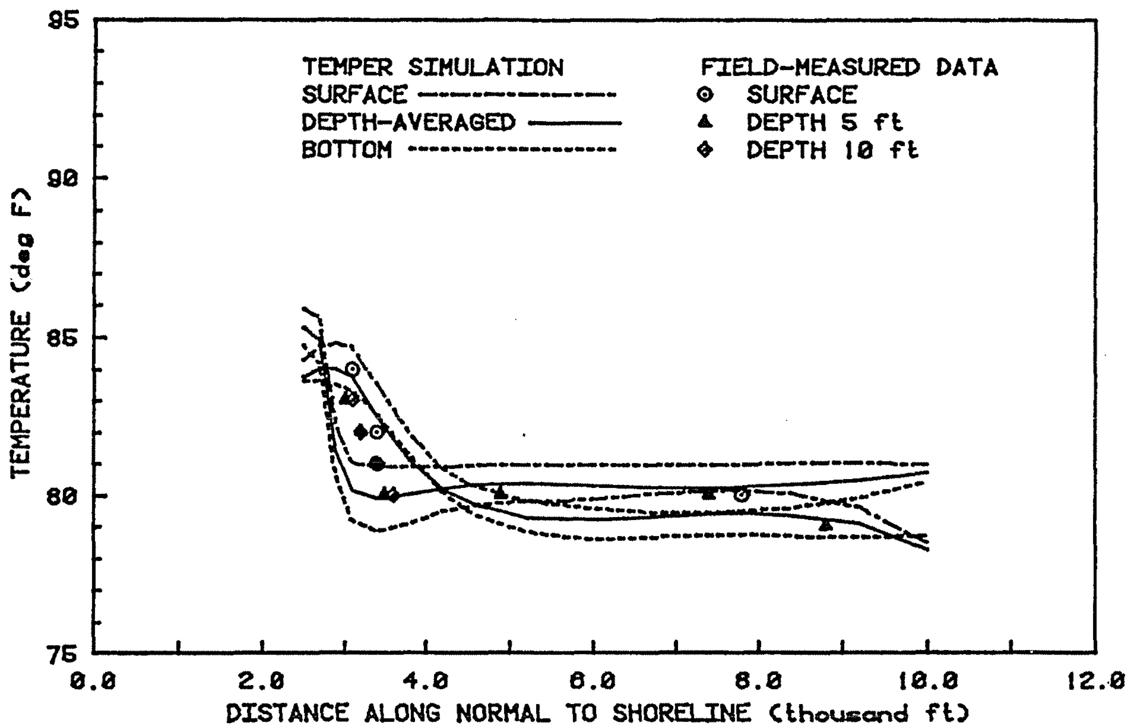


Figure 11.4 Comparison of the computer simulation results and the field-measured data for the temperature conditions along a transect at 1200 ft downstream from the discharge location of the Peach Bottom Atomic Power Station in the Conowingo Pond Reservoir from 8 a.m. to 1 p.m. on July 18, 1974.

(To convert ft to m, multiply by .3048.)

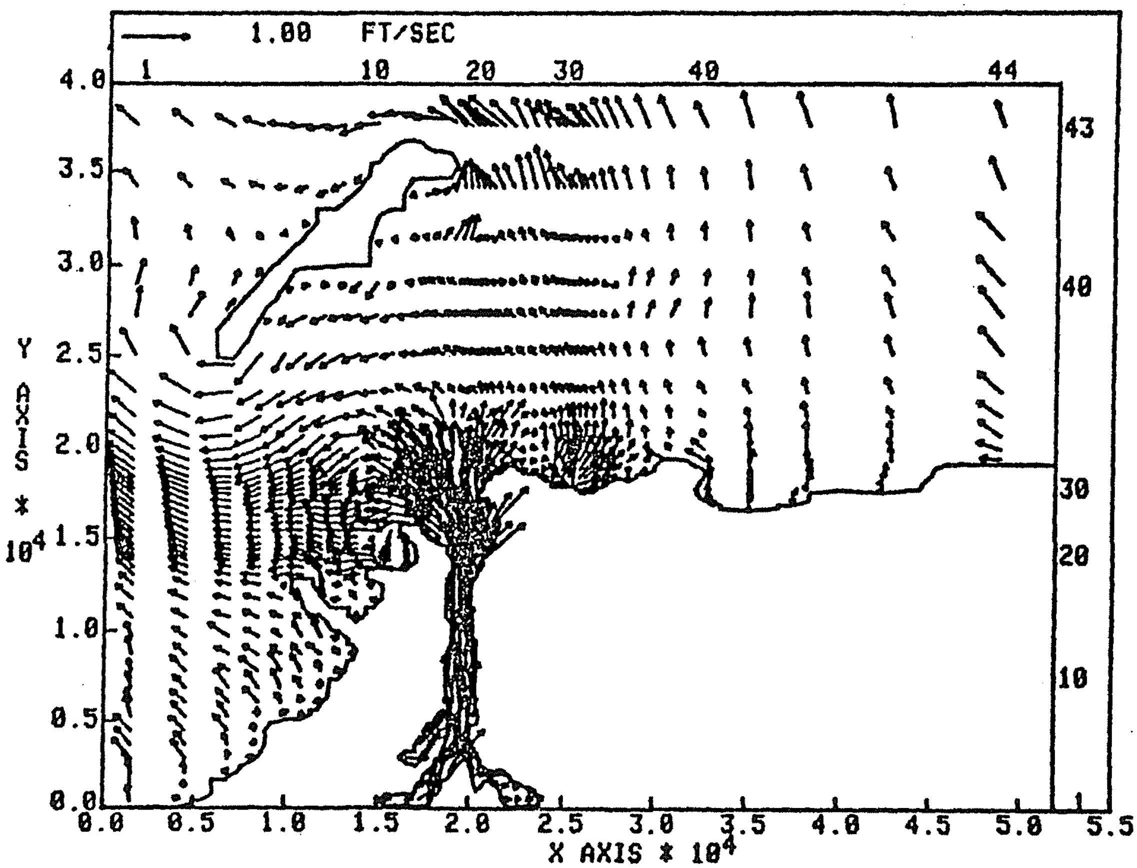


Figure 11.5 Computer simulation results for the two-dimensional depth-averaged flow velocity conditions in the Anclote Anchorage region for the actual Unit 1 operation of the Anclote Power Plant at 3 p.m. on June 25, 1975, during tidal stage: approximate maximum ebb.

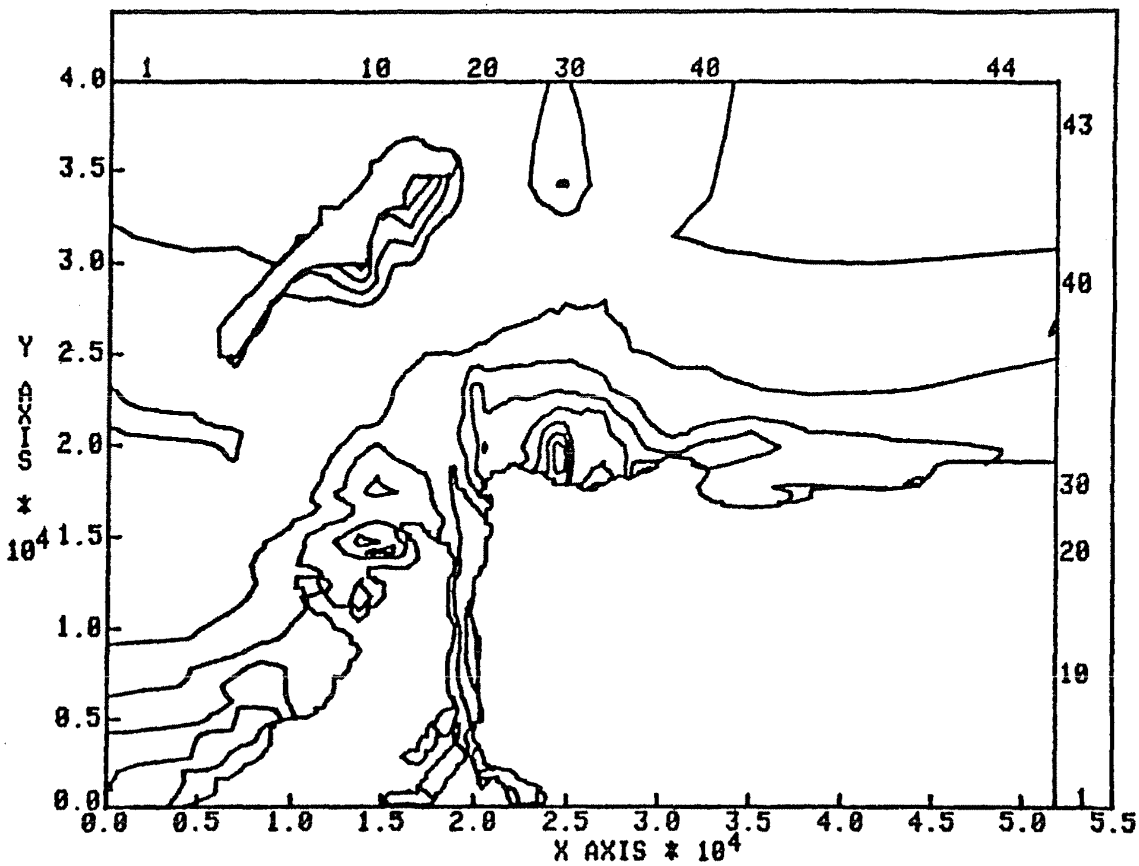


Figure 11.6 Computer simulation results for the two-dimensional depth-averaged water temperature conditions (isotherms with 1 F (1/1.8 C) gradation between minimum 84 F(28.9 C) and maximum 92 F(33.3 C)in the Anclote Anchorage region for the actual Unit 1 operational condition of the Anclote Power Plant at 3 p.m. on June 25, 1976, during tidal stage: approximate maximum ebb.

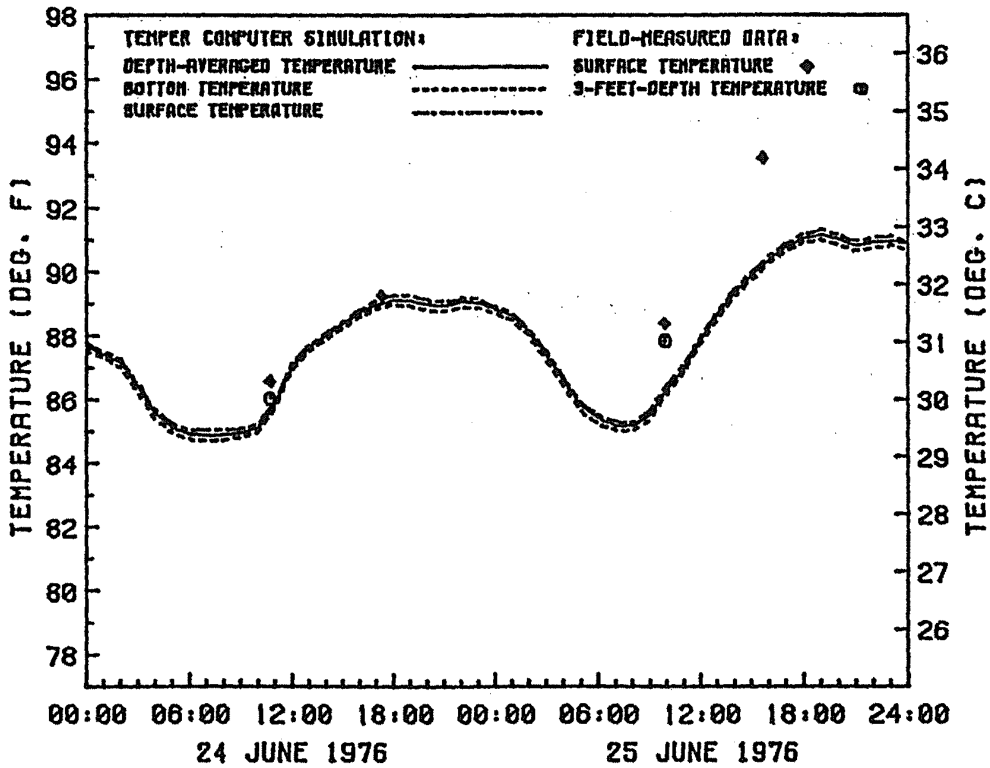


Figure 11.7 Comparison of the computer simulation results for the water temperature conditions (as continuous hourly variations) and the available field-measured water temperature data (intermittent) in the Anclote Anchorage region during the 2-day period June 24-25, 1976, at the field-sampling station 25.

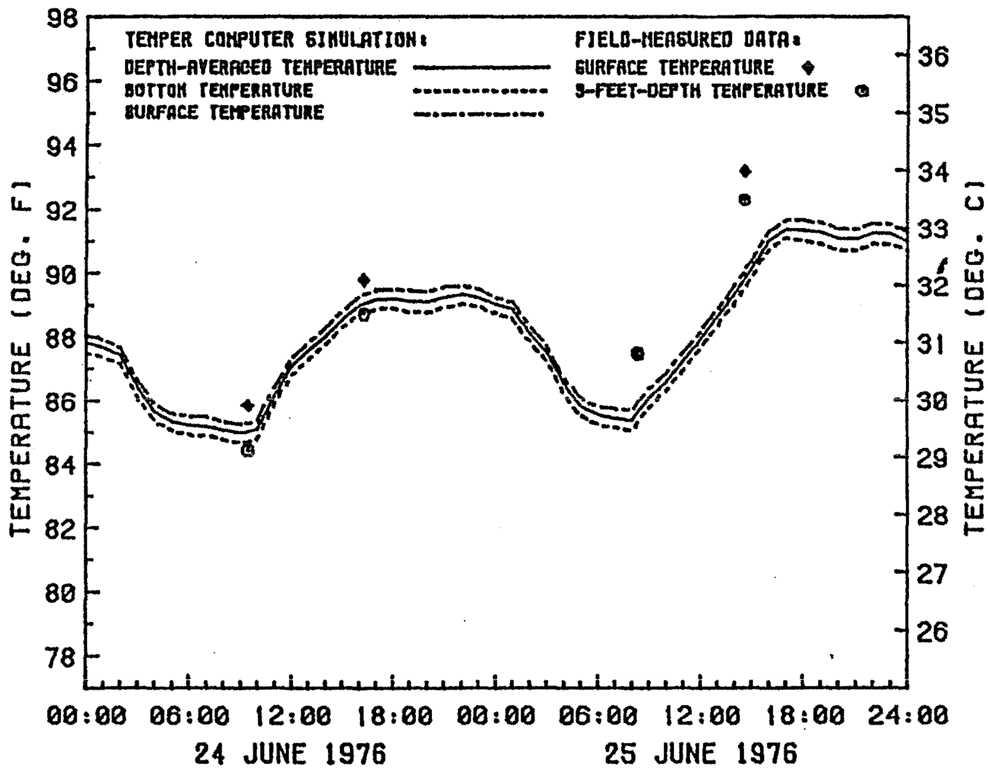


Figure 11.8 Comparison of the computer simulation results for the water temperature conditions (as continuous hourly variations) and the available field-measured water temperature data (intermittent) in the Anclote Anchorage region during the 2-day period June 24-25 1976, at the field-sampling station 1.

11.7 THERMAL EFFECTS (SCE, A-14, A-24/A-25)

It is true that the ambient temperature data supplied by the applicant (Attachment T) indicate the presence of lower ambient temperatures, at both the surface and the bottom, within the region of the San Onofre Kelp Bed than those predicted by the model. Typically, the discrepancy between the maximum values reported by the applicants and from the thermal model (for both surface and bottom) during the late summer is on the order of 4-5°C (7-9°F). However, the recently supplied values are for a period of record which spans only a few years (1975-1978). Thus, it is not possible to know if some of these data were collected during a period of time in which the waters are naturally warmer than long-term average. In evaluating the potential impacts of a long-lived facility such as SONGS (which will operate for up to 30 years), a worst-case evaluation is usually made to determine the effects which might occur under extreme conditions. Without knowledge of the relationship between the data supplied by the applicant and worst-case temperature conditions, these data cannot be relied upon for an assessment of the impacts on the kelp.

If the assumption is made that these data do include an interval in which there are temperatures which are warmer than usual (e.g., the warmest values reported in the last 15-20 years), then the conclusions of the applicant regarding the effects on kelp are probably correct. That is, the incremental temperature increase due to the operation of SONGS will not result in an adverse impact to the community, even under worse-case conditions.

The staff does not concur that the bottom water will necessarily remain unaffected.

The staff does not feel that the ambient temperature data available are complete enough for use in a worst-case analysis. Because such an analysis (based on the temperature data generated by the model) concluded that the impact to the benthic community will be insignificant (overall), it is clear that the effect under average conditions will likewise be insignificant. If the model does predict maximum temperatures which are unrealistically high, the conclusions based on such data provide additional assurances that the impacts will be negligible.

11.8 BIOLOGICAL RESOURCES (DOC, A-8/A-9; SCE, A-25/A-30; RJW, A-47)

The statement was not meant to imply that the only relationship between kelp beds and the commercial fishes is through the kelp detritus. Certainly, several of the commercially important species inhabit the beds (at least part of the time) for food-seeking activities and refuge. The paragraph was intended to portray, in brief, the ecological and commercial importance of kelp beds. Additional information on this subject is contained in the FES-CP.

Any revision to station design or operation for the express purpose of mitigating nonradiological impact to aquatic biota will be accomplished through procedures under the NPDES Permit Program administered by the California Water Quality Control Board. NRC is working closely with this Board and with other State resource agencies in the review of monitoring programs.

The cost-benefit analysis for the potential loss of biological resources due to the operation of SONGS 2 and 3 is addressed in the SONGS FES-CP, March 1973, Sections 13.2.4 and 13.4, and in Section 10 of this FES.

Technically, the statement is still correct. Although man-induced thermal effects may not be documented, their occurrence is likely in some areas.

It is true that the association of decreased kelp "health" with turbidity was made in conjunction with observations on the effects of sewage outfalls. However, the effect of turbidity on light attenuation is not a function of the source or type of the turbidity, except that a certain type of turbidity may not produce the same light reduction, at a given concentration, as another type of turbidity (e.g., as in the case of fine sand particles vs. suspended clay). Reference 15 includes a statement on the reduction of kelp "health" as a function of reduced available light for photosynthesis. It is reasonable to assume, therefore, that kelp "health" can be affected by many different types of turbidity.

It is true that nutrient depletion has been implicated in kelp canopy deterioration. It is also true that the exact mechanisms of kelp deterioration are not well known. Most studies have attributed cause-effect relationships between temperature and kelp demise, but the operating factors may well be complex, and may involve, for example, synergisms between several factors. To reflect this uncertainty, the text has been changed as per the suggested revision.

The study cited (number 13) gives one indication of the importance of kelp beds to fish communities. The relative importance of kelp for fish rearing, refuge, etc. may be disputed. However, for purposes of a worst-case analysis, it must be assumed that a more conservative interpretation is correct. It should be noted that the subject paragraph does not state that kelp beds are the most important fish habitat. Rather, it indicates that the community does have some unspecified importance.

The ecotypic variability of kelp temperature to tolerance is not well documented, though it may occur. Without definite knowledge, the more conservative values should be used in a worst-case analysis.

As stated earlier (see response to SCE, A-2), the ambient temperature data supplied by the applicants do not appear to be entirely adequate for an analysis.

Increased survace nutrient levels from outfall induced upwelling have not been adequately demonstrated, though the phenomenon may occur.

The discharge for SONGS will be more or less continuous (except during shutdowns and power reductions), although it is true that when the plume reaches the kelp bed it is not likely to impinge the bed for a long period of time. This is acknowledged in the text and is one major reason why the conclusion was reached that the effect on the kelp bed is not likely to be severe (p. 5-27, paragraph 5). To avoid ambiguity, the subject statement has been revised.

The reference to Sect. 5.3.1 as the basis for the 19°C figure has been deleted. It is, however, based on results of the staff's thermal model. If this temperature represents the extreme high end which occurs during an average year, then it may well represent a temperature which could occur over an extended period of time during a "warm year." Because the values from the model are conservative, the word "typical" has been deleted from that sentence.

The staff's thermal analysis (Sect. 5.3.1) does not support the conclusion that increases in bottom temperatures are unlikely. The statement remains valid, even if the turbid water is discontinuous and relatively low in suspended solids; i.e., the presence of any increased turbidity will add to the probability of detrimental effects occurring to the kelp bed.

The analysis of the effect of the operation of SONGS on the closest kelp bed (San Onofre) was based on the latest thermal-hydraulic predictions made by our staff. The staff agrees that if this assessment of temperature configurations proves to be inaccurate that the analysis of kelp bed effects would have to be reassessed.

The granting of a 316(a) exception for normal operation, if such is needed, will not affect the operating characteristics of the facility; thus, the pedicted impacts remain valid. If a 316(a) type process becomes required by the State which results in operational changes, the impacts of that altered operating mode will have to be assessed when such conditions are known.

11.9 WATER QUALITY (EPA, A-38/A-40)

Section 5.3.1.1 and Fig. 5.3 are based on the applicant's thermal analysis. Section 5.3.1.2 is a description of the staff's thermal analysis. This analysis includes all three units operating at 100% capacity and the ambient temperature is defined as the water temperature in the absence of all units. The possibility of recirculation among all units is an integral part of the staff's model. The results described in Section 5.3.1.2 address all aspects of the State thermal standards including excess temperatures at the surface of ocean substrates.

Additional information on the effects of the operation of the facility are found in the FES-CP, although in some cases the modification of operational characteristics since the CP stage has necessitated a reevaluation of the impacts. Such reanalyses are found in this document.

In the staff's opinion, the two documents (the FES-CP and FES-OL) provide "state-of-the-art" evaluations of how the acquatic ecosystem will be affected. In most cases, too little information is available to quantitatively predict the areal extent of an effect on a given species. For this reason, operational monitoring programs are required which are designed to detect significant changes which may occur so that mitigation can be instituted.

The terms "minimal," "acceptable," and "not significant" relate to a judgment made regarding the predicted impacts of the facility on the environment. A possible effect is termed insignificant if, for example, the impact is predicted to only occur locally in a nonunique, or widespread, population community of organisms, etc. When it is not possible to reach a firm conclusion regarding the significance of a projected impact (even under worst-case conditions), mitigation is either recommended immediately or an extensive monitoring program is stipulated.

The study report concerning use of the heat treatment process addresses a matter beyond the NRC's purview in accordance with the Federal Water Pollution Control Act Amendments of 1972. Thus, while the NRC must evaluate and consider the environmental impacts attributable to use of such heat treatment process, such consideration is limited to a determination of the impacts and their significance in terms of the cost-benefit analysis for this facility; any changes in the system or its use must be directed by the California Regional Water Quality Board and/or the Environmental Protection Agency. The applicant will provide the study report directly to those agencies, as well as to the NRC, when available.

For the foregoing reasons, we do not believe that the report itself is an integral part of the Draft Environmental Statement. Of course, as noted above, the NRC will consider the impacts attributable to the heat treatment process in the Final Environmental Statement as stated in Section 5.4.2.1. In this connection, the staff considers it to be no different than any other report of a study or analysis performed by a license applicant in support of its application. The staff is aware of no legal requirement which would give the report independent status such as EPA suggests, in the context of the NRC's licensing review. The status of this report in terms of the determination to be made under Section 316(a) of the FWPCA is a matter left to that agency charged with making that determination.

Sect. 5.4.2.2 concludes that significant impacts are unlikely, even under worst-case conditions. The effluent characteristics of SONGS must conform to the State standards prevailing at the time of the operation of the facility.

It is not the purpose of the staff analysis, as provided in the DES, to make rulings regarding statutory requirements, but rather to assess the impacts of proposed operation. In making this analysis it is not assumed that standards will be satisfied and, therefore, the environmental consequences of any violations resulting from the proposed plant operation is inherent in the staff's conclusions.

11.10 NEED FOR PLANT (MIL, A-45)

Table 2.2 of the FES relates to projected population growth within 16 km of the San Onofre site for the period 1976 to 2020.

Tables 8.3 and 8.4 are related to the electrical growth projected within the service areas of Southern California Edison Company and San Diego Gas and Electric Company for the periods 1976-1985. The combined annual growth rates for peak demand and energy for this period is 4.3 and 4.6%, respectively.

Population within 16 km of the site does not necessarily reflect electrical growth in the applicant's service areas.

11.11 SEISMOLOGY (EPA, A-40; MIL, A-45; RJW, A-49)

The staff's review and evaluation of the geological and seismological aspects of the San Onofre Nuclear Generating Station Units 2 and 3 is presented in the staff's Safety Evaluation Report, published December 1980. Included in the SER is a discussion of the potential for and nature of seismic activity at the site and its vicinity as well as of the design and construction measures taken by the applicants to prevent damage to the facility and its component parts. The staff considers that its assessment of the environmental impact of postulated accidents presented in Chapter 7 adequately accounts for the consequences of any accident caused by seismic activity. This chapter discusses the consequences of accidents regardless of cause.

Regarding the potential for affecting water quality and for offsite radiological contamination, to the extent such impacts are the result of airborne transportation of radionuclides, the consequences are included in the discussion in Chapter 7. The liquid pathway, because of the hydrological environment at the site, does not present a viable transport mechanism which could impact water quality or would otherwise result in offsite radiological consequences.

11.12 URANIUM PRICES (RJW, A-49)

The cost-benefit analysis in the DES is based on 1976 data, at which time the price of uranium was \$18.10/kg (\$40/lb) U_3O_8. Presuming SCE used the then existing U_3O_8 price in their cost-benefit analysis for SONGS 2 and 3, they would be using a price that reflected the rapid increase in prices in the 1973-1976 period. To extrapolate future prices on the basis of the 1973-1976 price increase would be erroneous in that uranium prices decreased 9% in real terms during 1977. Thus, it is inappropriate to consider a price escalation which is not even valid for a 5-year period of the uranium market for a cost-benefit analysis which covers the 30-year lifetime of a reactor. It would be just as appropriate (or inappropriate) to extrapolate the recent 9% decrease in uranium prices for use in the analysis. Many factors must be carefully investigated to estimate future uranium prices, and simplistic methods cannot be justified.

Long-term contracts are not generally tied to market prices at time of delivery or a 7% per year escalation, whichever is greater, based on current data. In fact, most long-term uranium requirements are not under contract, so it is inappropriate to make any generalization about the nature and terms of those contracts. Even if future contracts were based on the greater of market price or 7%/year escalation, it does not follow that fuel costs will increase to prohibitively high levels. If future prices were related to market prices and market prices do not increase substantially (it has not been established that they will), then the uranium cost component of fuel costs would not increase very much. And, if prices increase at 7%/year, they would probably just be keeping pace with inflation and thus not be relevant to a constant dollar analysis.

The cost of purchasing uranium is only one component of nuclear fuel costs, the other being, for example, separative work, UF_6 conversion, and fabrication. Thus, overall nuclear fuel costs would not escalate in proportion to the increase in uranium prices.

11.13 ACCIDENTS (HEW, A-10; MIL, A-45; RJW, A-49)

These comments were addressed to the Accident Section (Section 7) published in the Draft Environment Statement (DES), dated November 1978. In January 1981, the staff revised Section 7 and issued it for comment as a supplement to the DES. The January 1981 Supplement is included as Section 7 of this Final Environmental Statement (FES). The staff believes FES Section 7 is responsive to those accident comments previously addressed to the DES.

(FHA, A-53)

Part 50.13 of 10 CFR does not require a licensee "to provide for design features or other measures for the specific purpose of protection against the effects of (a) attacks...by an enemy of the United States...or (b) use or deployment of weapons incident to U.S. defense activities." The staff recognizes that acts of war would likely produce severe environmental impacts wherever they might take place.

(RJW, A-56, A-59)

The supplement is based on site-specific data, as described in Section 7.1.4.2. While not specifically stated in the supplement, U.S. average, year 2000 estimated, population data were used beyond 560 km (350 miles). The site-specific meteorological data used included lid heights to account for vertical mixing characteristics.

(RJW, A-58)

Both the staff and SAI used very similar methodologies in their analysis, and they both represent improvements over the Reactor Safety Study. There are some differences, however, in assumptions and data used in each study that lead to the variabilities or uncertainties that are inherent in such calculations. These differences appear in:

- accident release characteristics - probabilities and magnitudes;
- emergency response assumptions;
- meteorological data; and
- demographic data.

Specific consequence values that commentors quote from the SAI-OES report cannot be directly compared with those reported in the staff's draft supplement since the former are not associated with specified probability levels while the latter are. The staff has not made a

detailed comparison of the results of the two reports but judges that they are in agreement within the estimated bounds of uncertainties and assumptions associated with the current state of probabilistic risk assessments.

(RJW, A-58, A-59, A-60)

The studies of the San Onofre site relative to earthquake potential is extensively discussed in Section 2.5 of the Safety Evaluation Report (NUREG-0712, December 1980). The staff's position is that the safe shutdown earthquake is correctly determined for this site. A discussion of natural phenomena as initiators of accidents has been added to Section 7.1.4.2.

(RJW, A-58, A-59)

If Unit 1 had a meltdown, the staff agrees that it would impact the operation of Units 2 and 3. However, the ability to shut down both units following an accident at Unit 1 would not be impaired.

(RJW, A-59)

The reactor vessel was installed with its reference mark 180 degrees from the desired location. As discussed in the Safety Evaluation Report (Section 5.3.4), the flow skirt, which is not symmetrical, was installed in the direction to agree with the vessel's orientation and procedures for fuel handling, which reference the vessel orientation, were modified. No rewiring was necessary as a result of the error.

(RJW, A-59)

The dewatering well cavities were discussed in the Safety Evaluation Report in Section 2.5. It was determined that there would be no impact on seismic Category I structures.

(RJW, A-59)

The beach visitors were specifically addressed (e.g., Sections 7.1.3.2 and 7.1.4.6 and Table 7.1.4-5).

(RJW, A-60)

The staff has concluded that acts of sabotage, as initiating events, do not contribute significantly to the probability of accidents due to the Commission's safeguards requirements. Section 7.1.4.2 has been modified to discuss this point.

(RJW, A-60)

While it is true that one-half of the population of the State of California lies within 160 km (100 miles) of the San Onofre site, the staff does not consider this to be a relevant observation. The staff's focus on demographic data for site suitability and site comparison purposes has been traditionally within 80 km (50 miles) of plant sites.

The discussion in Section 7.1.4.3 indicates that most of the accident impacts occur within 50 miles of the site. The staff has compared the total population within 50 miles of the site with the total population within 80 km (50 miles) of other nuclear plant sites and has found that San Onofre does not have a uniquely large population. Moreover, it is important to note that, as stated in Section 7.1.4.2, the site-specific population projected to the year 2000 both in magnitude and distribution has been used in the calculations through all regions to 160 km (100 miles) and beyond. Those fractions of the consequences which take place up to 16, 48, 80, 160 km (10, 30, 50, 100 miles) or beyond, are accounted for in the results presented. The site does not have a uniquely large population contained within any of the above mentioned distances.

(RJW, A-60)

The San Onofre Units 2 and 3 at 3390 MWt are typical of the upcoming generation of reactors. The power level of each plant was specifically used in determining the inventory of the core for the risk calculations. Salem 1 is presently operating at a comparable power level of 3338 MWt.

(RJW, A-60)

The production of farm and dairy products is specifically considered in the calculation. Differences among the states (and countries) potentially impacted are taken into account.

(RJW, A-60)

The impacts within the Mexican borders were included in the evaluation. The method of determination of data for Mexican agricultural products is discussed in Section 7.1.4.2. Although not explicitly stated, the population within the Mexican border was included on a site-specific basis out to 560 km (350 miles) from San Onofre.

(RJW, A-61)

The staff recognizes the potential efficacy of drugs in mitigating consequences of offsite exposures. However, in the case of potassium salts of stable iodine, the staff does not require provision for distribution to the public.

(RJW, A-61)

Section 7.1.4.4 discusses that the condemning of foodstuffs was specifically considered and the interdiction of contaminated property "...until it is either free of contamination or can be economically decontaminated" was assumed.

(RJW, A-61)

The subject of filtered venting systems for the containment is being addressed in rulemaking, as discussed in NUREG-0660, 2.B.8. The whole subject of TMI-2-related improvements and the fact that no credit would be taken for them is discussed in the last paragraph of the section cited.

(RJW, A-61)

It is the staff's position that such a "worst case accident" is much too remote and speculative to require analysis under NEPA.

(UCS, A-63)

The staff believes that its treatment of a multiplicity of possible accident scenarios represents a reasonable and appropriate implementation of the guidance provided in the Commission's Statement of Interim Policy.

(UCS, A-63)

The probabilities of occurrence of releases in the nine categories are explicitly given in Table 7.3 and the probabilities of occurrence of specific levels of environmental consequences are given in Table 7.4. The staff judges that this is within the intent of the quoted part of the sentence and the additional directive in the Interim Policy which states: "The environmental consequences of releases whose probability of occurrence has been estimated shall also be discussed in probabilistic terms." See also the staff's answer to Joint Intervenors RJW, A-58.

(UCS, A-64)

The staff has presented a measure of individual risk in Figures 7.7 and 7.8. Table 7.4 and its associated figures and Table 7.5 provide group information. The discussion of relative susceptibility of various sub-groups of the population is given in Section 7.1.1.3. The staff judges that this conforms to the further directive that the discussion be "...in a manner that fairly reflects the current state of knowledge regarding such risks."

(EPA, A-66)

The Supplement is a replacement for Chapter 7 in the existing Draft Environmental Statement for the Operating License stage (November 1978). It is not a replacement for the accident sections of the Construction Permit stage Environmental Statements of 1973.

(EPA, A-66)

Nine tables could have been provided to show the impact contributions of the nine categories. It is the staff's judgment, however, that the summary table, reflecting sums of the contributions from all of the categories, is sufficient. Information regarding the relative contributions of the release categories is available in the Reactor Safety Study, WASH-1400.

(EPA, A-66)

The Design Basis Accidents are included because they are used in the Safety Evaluation Report to assess the adequacy of the performance of certain engineered safety features. In the SER, the DBAs are compared to the suitably small fractions of 10 CFR 100 for those accidents that are considered likely (infrequent accidents).

(EPA, A-66)

The DBAs are judged not to be significant contributors to environmental risks and have not been subjected to the same kind of probabilistic analysis as the more severe accidents that are treated.

(EPA, A-66)

The staff believes that it is more informative to discuss the environmental risks associated with accidents separated from those attributable to normal operations. Both may be found in the Final Environmental Statement. Risks associated with the operation of both Units 2 and 3 are, to a first approximation, the sum of the risks associated with each unit individually.

(EPA, A-67)

We agree certain biological changes in children and adults may be detected occasionally at doses as low as 10 rem (e.g., slight, temporary reductions in circulating lymphocytes). However, at doses of 25 rem or greater, such effects become measurable in nearly all exposed persons. In addition, although such changes have no physiologically significant impact, they can be clinically measured. We selected 25 rem as a point above which potentially serious effects due to radiation exposure (e.g., prodromal vomiting) become apparent to physicians and a point below which no difference between exposed and unexposed populations is apparent in terms of latent cancer incidence.

(EPA, A-67)

The NRC State Liaison Officer has informed us that the Region IX RAC has completed its review of the local plans for the environs of San Onofre. The licensee has transmitted to the NRC copies of emergency plans for the following:

San Onofre, San Clemente and Daheny State Park and Beach Areas

San Juan Capistrano City

Camp Pendleton

Orange County

Unified San Diego County

Interagency Agreement between San Diego County, Orange County, City of San Clemente, City of San Juan Capistrano; Marine Corps, Camp Pendleton; State Department of Parks, Pendleton Coast Area.

The staff preliminary review of these documents affirms its judgment that the plans are, in fact, in an advanced stage of development even though they have not been submitted for formal review. A full-scale exercise for the San Onofre site and its environs is scheduled for May 13, 1981.

(EPA, A-67)

The NRC staff Safety Evaluation Report, dated February 1981, states that the San Onofre onsite emergency plan, when revised in accordance with the applicant's commitments, will provide an adequate planning basis for an acceptable state of emergency preparedness, and will meet the requirements of 10 CFR Part 50 and Appendix E, thereto. This is still the staff's conclusion.

The SER states that the plan must be revised to address the final criteria and implementation schedule for the emergency centers and their functions, emergency manpower levels, and upgraded meteorological program, and to address the impact of earthquakes on emergency plans for the site and its environs. The NRC staff position is that the emergency plans are sufficiently complete to justify the estimates of parameters used in the consequence model.

It is true that the State of California does not use the U.S. EPA Protective Action Guides (PAGs). The State of California has elected to base its Protective Action Guides on the concept that no member of the general public should receive more than 500 millirem per year. The emergency plans of the local authorities in the environs of the San Onofre plant have followed the State's guidance. This guidance is more conservative than the EPA guidance, i.e., protective actions would be recommended at a lower projected dose. Consequently, it is reasonable to expect that if protective actions were to be taken at a lower value of projected dose, then exposures would be reduced.

(EPA, A-67)

The figure has been revised to present a more meaningful directional risk. The meaning of the new figures is discussed in Section 7.1.4.6. The scale of the figures has been expanded (a smaller distance from the plant shown) and it has been redrawn in an attempt to improve legibility.

(EPA, A-67)

Standard methods for estimating costs of reactor building cleanup and decontamination and replacement power for the economic risk calculations are under development. Reasonable estimates of plant decontamination and replacement power have been made, however, and are discussed in Section 7.1.4.6. Staff conclusions on the benefit cost balance are reported in Section 10 of the FES.

(SCE, A-68)

It is clearly stated in the third paragraph of Section 7.1.4.1 that the evaluations of the limiting faults and infrequent accidents are used to implement the provisions of 10 CFR 100. The conclusions regarding siting are in the Safety Evaluation Report and its supplements.

(SCE, A-69)

Section 7.1.4.2 states that the estimates of the consequences may have as large error bounds as for the probabilities. Any uncertainty in the fractions of nuclides released contributes to the error bounds on the consequences, as well as uncertainties in the meteorological and health effects models. The subject of releases of certain nuclides, mainly the radioiodines, is presently under review by the staff.

APPENDIX A

COMMENTS ON DRAFT ENVIRONMENTAL STATEMENT

APPENDIX A

COMMENTS ON DRAFT ENVIRONMENTAL STATEMENT

A-1

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UNITED STATES DEPARTMENT OF AGRICULTURE
SCIENCE AND EDUCATION ADMINISTRATION

OFFICE OF THE DEPUTY DIRECTOR FOR
AGRICULTURAL RESEARCH
WASHINGTON, D.C. 20250

Subject: NRC Draft Environmental Statement

To: William H. Regan, Jr.
U.S. Nuclear Regulatory Commission
Environmental Projects Branch 2
Division of Site Safety and Env. Analysis
Washington, D.C. 20555

We have reviewed the draft environmental impact statement entitled San Onofre Nuclear Generating Station, Units 2 and 3, Southern California Edison Company, San Diego Gas & Electric Company, dated November 1978.

We have no comments to add to the evaluation provided by your staff. We do appreciate the opportunity of reviewing this statement.

H. L. BARROWS
Acting Deputy Assistant Administrator

A-2

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U.S. DEPARTMENT OF AGRICULTURE
ECONOMICS, STATISTICS, and COOPERATIVES SERVICE
WASHINGTON, D.C. 20250

December 8, 1978

SUBJECT: Draft Environmental Statement

TO: William H. Regan, Jr., Chief
Environmental Projects Branch 2
Division of Site Safety and
Environmental Analysis
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

We have no comments on the Draft Environmental Statement related to operation of San Onofre Nuclear Generating Station, Units 2 and 3 by Southern California Edison and San Diego Gas and Electric Companies.

Melvin L. Cotner
MELVIN L. COTNER
Director
Natural Resource Economics Division

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50-362 A

DEPARTMENT OF HOUSING AND URBAN DEVELOPMENT
AREA OFFICE
2500 WILSHIRE BOULEVARD, LOS ANGELES, CALIFORNIA 90057

REGION IX
450 Golden Gate Avenue
P.O. Box 36003
San Francisco, California 94102

December 19, 1978

IN REPLY REFER TO:

U.S. Nuclear Regulatory Commission
Attention: Director, Division of Site Safety
and Environmental Analysis
Washington, D.C. 20555

Gentlemen:

Subject: San Onofre Nuclear Generating Station
Units 2 and 3
Draft Environmental Statement
Docket Nos. 50-361 and 50-362

We have reviewed the captioned document and have no comments to offer on it. There is no need to send this office a copy of the Final Environmental Statement.

Sincerely,

Roland E. Camfield, Jr.
Area Manager

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A-3

DEPARTMENT OF THE ARMY
LOS ANGELES DISTRICT, CORPS OF ENGINEERS
P. O. BOX 2711
LOS ANGELES, CALIFORNIA 90053

50-361P
3620

SPLED-E

2 January 1979

Mr. Wm. H. Regan, Jr., Chief
Environmental Projects Branch 2
Division of Site Safety and Environmental Analysis
United States Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Mr. Regan:

This is in response to a letter from your office dated 30 November 1978 which requested review and comments on the Draft Environmental Impact Statement for the San Onofre Generating Station, Units 2 and 3, proposed by Southern California Edison Company and San Diego Gas and Electric Company.

The proposed plan does not conflict with existing or authorized plans of the Corps of Engineers. We have no comments on the environmental statement for the proposed action.

Thank you for the opportunity to review and comment on this statement.

Sincerely yours,

NORMAN ARNO
Chief, Engineering Division

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UNITED STATES DEPARTMENT OF AGRICULTURE
SOIL CONSERVATION SERVICE
2828 Chiles Road, Davis, CA 95616

January 9, 1979

William H. Regan, Jr., Chief
Environmental Projects Branch 2
Division of Site Safety and
Environmental Analysis
United States Nuclear Regulatory
Commission
Washington, D. C. 20555

Docket No.: 50-361
and 50-362

Dear Mr. Regan:

We acknowledge receipt of the draft environmental statement for San Onofre Nuclear Generating Station, Units 2 and 3, Southern California Edison Company, San Diego Gas & Electric Company in San Diego County, California, that was addressed to USDA Soil Conservation Service on November 30, 1978, for review and comment.

We have reviewed the above draft and have the following comments.

1) Provisions for erosion control and water management during construction as well as conservation treatment of disturbed areas following construction were inadequately addressed. We suggest that an erosion control plan be developed to adequately address the erosion hazard both during and following construction.

2) Approximately 10 acres of prime land will be lost to access roads and transmission towers. Mitigation or projected impacts from this loss were not adequately discussed. We suggest further discussion in the statement to address the prime land issue.

We appreciate the opportunity to review and comment on this proposed project.

Sincerely,

Francis C.H. Lum

FRANCIS C. H. LUM
State Conservationist

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FEDERAL ENERGY REGULATORY COMMISSION
WASHINGTON, D.C. 20426

January 17, 1979

IN REPLY REFER TO:

Mr. William H. Regan
Division of Site Safety and
Environmental Analysis
Nuclear Regulatory Commission
Washington, D. C. 20555

Dear Mr. Regan:

I am replying to your request of November 30, 1978 to the Federal Energy Regulatory Commission for comments on the Draft Environmental Impact Statement for the San Onofre Nuclear Station Units 2 and 3, California. This Draft EIS has been reviewed by appropriate FERC Staff components upon whose independent evaluation this response is based.

The staff concentrates its review of other agencies' environmental impact statements basically on those areas of the electric power, natural gas, and oil pipeline industries for which the Commission has jurisdiction by law, or where staff has special expertise in evaluating environmental impacts involved with the proposed action. It does not appear that there would be any significant impacts in these areas of concern nor serious conflicts with this agency's responsibilities should this action be undertaken.

Thank you for the opportunity to review this statement.

Sincerely,

Jack M. Heinemann

Jack M. Heinemann
Advisor on Environmental Quality

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United States Department of the Interior

OFFICE OF THE SECRETARY
WASHINGTON, D.C. 20240

JAN 1 6 1979

In Reply Refer To:
ER 78/1161

Mr. William H. Regan, Jr.
Chief, Environmental Projects Branch 2
Division of Site Safety and
Environmental Analysis
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Mr. Regan:

The Department of the Interior has completed its review of the draft environmental statement for San Onofre Nuclear Generating Station Units 2 and 3. We have comments in only two areas of our jurisdictional concern as set forth below.

A-5

Recreation Resources

The discussion of recreation impacts states that restrictive use of the beach area was unanticipated at the time issuance of the construction permit was being considered. Since no explanation is given, it is unclear to us how such a significant impact, the loss of recreational and scenic open space, could have been overlooked during the earlier planning stages. The final statement should disclose the reasons which now require restrictions upon beach use.

Although there is now recognition of the impact, we see no attempt being made by the applicant to mitigate the loss of recreation space and opportunity. With respect to the scenic quality of the area, we find the intended plan to construct an eight foot chain-link fence extending over three-fourths of a mile along the beach quite objectionable. Design of the walkway deserves much more attention. Given the fact that this stretch of beach is rather removed from the developed portions of the state park units and therefore receives minimal use and given the scenic nature of the beach area and bluffs it would certainly seem more preferable and perhaps sufficient to consider posting the area as to its restrictive use. If a barrier is still needed, a more aesthetically sensitive, light railing may best fulfill the need to restrict access.

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Cultural Resources

We are pleased that the NRC staff has directed the applicant to consider historic, archeological, and Native American cultural resources in its planning process. We understand that existing and possible new transmission corridors will be surveyed for such resources. However, we strongly urge that the applicant allow enough flexibility in its planning to actually take the results of these surveys into account in its final placement of tower bases, access roads, and proposed substations. This would include allowing the State Historic Preservation Officer enough time to properly evaluate the surveys results and make appropriate recommendations. In addition, any new land used for material storage or other project activities outside the transmission corridors should also be checked for cultural resources.

We hope these comments will assist your efforts in preparing the final environmental statement.

Sincerely,

Larry E. Meierotto
Deputy Assistant SECRETARY

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362

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January 19, 1979

United States Nuclear Regulatory Commission
Washington, D. C. 20555

Attention: Director, Division of Site, Safety and Environmental Analysis

Gentlemen:

Pursuant to the notice published in the Federal Register with respect to comments on the Draft Environmental Statement (DES), the Cities of Anaheim and Riverside, California wish to submit the following comments.

Anaheim and Riverside believe the Final Environmental Statement (FES) should be amended in Section 3, entitled "Need for the Station", to reflect probable ownership by the Cities of a portion of Southern California Edison Company (SCE), 80% interest in Units 2 and 3.

A-6

The Applicants and Anaheim and Riverside, entered into a Letter Agreement dated November 1, 1977 which incorporated other proposed Agreements, including a Participation Agreement which provides for Anaheim to acquire 1.66% of SCE's 80% interest in Units 2 and 3, and for Riverside to acquire 1.79% of Edison's 80% interest in Units 2 and 3. The Letter Agreement was entered into by the Parties because a question was raised as to whether SCE or SDG&E would lose the investment tax credit with respect to its ownership share of Units 2 and 3 due to Anaheim and Riverside, public agencies, owning an undivided interest in Units 2 and 3. The Letter Agreement further provides, however, that when this question is satisfactorily resolved in the opinion of each party to said Agreement, the Participation Agreement attached thereto will be executed by the Parties.

The Internal Revenue Service has issued Revenue Ruling 78-268, which holds that undivided ownership in property by exempt and non-exempt entities does not of itself disqualify the portion of the property owned by the non-exempt entity from taking investment tax credit with respect to its share of such property. Moreover, SCE and SDG&E received a private letter ruling, dated August 16, 1978 which holds with respect to San Onofre Nuclear Generating Station, Units 2 and 3, that SCE and SDG&E will not lose investment tax credit with respect to their undivided interest in Units 2 and 3 after the sale of the interest to Anaheim and Riverside. However, that Private Letter Ruling contained language which SCE and SDG&E believe to be ambiguous and therefore on October 27, 1978 they filed a Request for Clarification of the Private Letter Ruling of August 16, 1978. The Request for Clarification is still pending before the Internal Revenue Service, but we believe it will be favorably acted upon in the next several weeks.

Anaheim and Riverside are currently, and have for some years, been wholesale

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United States Nuclear Regulatory Commission
January 15, 1979

Page Two

customers of SCE. Anaheim and Riverside purchase all of their capacity, and most of their energy requirements from SCE. Anaheim and Riverside have agreements with Nevada Power Company wherein each City purchases economy non-firm energy from Nevada Power Company. These agreements will expire by their own terms in 1980. Anaheim and Riverside do not currently own any generating resources.

In 1978 Anaheim's peak demand was 388 megawatts. The estimated peak demand for 1978 was 394 megawatts. During 1978 Anaheim purchased two billion kilowatt hours of energy. For the period 1979 through 1990 it is estimated the peak demand for Anaheim will increase in differing amounts. The smallest amount of increase for electrical demand in any year during that period is estimated to be 3.1 percent and the highest amount of increase for any year 4.8 percent. It is also estimated for the same period of time that energy requirements for Anaheim will increase with the lowest estimated annual increase being 3.6 percent and the highest estimated annual increase being 5.0 percent.

In 1978 Riverside's peak demand was 278 megawatts. The estimated peak demand for 1978 was 260 megawatts. During 1978 Riverside purchased over one billion kilowatt hours of energy. For the period 1979 through 1990 it is estimated the peak demand for Riverside will increase with the smallest annual increase to be 4.0 percent and the highest annual increase to be 5.4 percent. It is also estimated for the same period of time that the energy requirements for Riverside will increase with the lowest annual increase to be 4.0 percent and the highest annual increase to be 5.4 percent.

The acquisition of an ownership interest in Units 2 and 3 by Anaheim and Riverside does not change the conclusion that the Units are needed to meet the electrical load served by SCE, Anaheim and Riverside. The load forecasts of SCE include the loads served by Anaheim and Riverside. Therefore, whether you include the loads of Anaheim and Riverside in the SCE forecast of loads or break them out and identify them separately, the need for the station is the same.

Very truly yours,

Alan R. Watts

ALAN R. WATTS
Special Counsel

Cities of Anaheim and Riverside

ARW:jm:D:11

cc: Attached Listing

A-7

Mr. John Bury
Southern California Edison Company
2244 Walnut Grove Avenue
P. O. Box 800
Rosemead, California 91770

Mr. Mark Medford
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Wm. H. Regan, Jr.
Environmental Projects Branch 2
Division of Site Safety and Environmental Analysis
United States Nuclear Regulatory Commission
Washington, D. C. 20555

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City of Riverside
3900 Main Street
Riverside, CA. 92522

7901310097

UNITED STATES DEPARTMENT OF COMMERCE
The Assistant Secretary for Science and Technology
Washington, D.C. 20230
(202) 377-~~3111~~ 4335

50-361
362

January 22, 1979

Director, Division of Site Safety
and Environmental Analysis
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Sir:

This is in reference to your draft environmental impact statement entitled "San Onofre Nuclear Generating Station, Units 2 and 3, Southern California Edison Company, San Diego Gas & Electric Company." The enclosed comments from the National Oceanic and Atmospheric Administration are forwarded for your consideration.

Thank you for giving us an opportunity to provide these comments, which we hope will be of assistance to you.

We would appreciate receiving 10 copies of the final statement.

Sincerely,

Sidney R. Galler
Sidney R. Galler
Deputy Assistant Secretary
for Environmental Affairs

Enclosures from: Gordon G. Lill--National Ocean Survey
Gerald V. Howard--National Marine Fisheries Serv

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UNITED STATES DEPARTMENT OF COMMERCE
National Oceanic and Atmospheric Administration
NATIONAL OCEAN SURVEY
Rockville, MD 20852

OA/C52x6

JAN 10 1979

TO: PP - Richard L. Lehman

FROM: OA/Cx1 - Gordon G. Lill

SUBJECT: DEIS #7812.06 - San Onofre Nuclear Generating Station, Units 2 and 3

The subject statement has been reviewed within the areas of NOS responsibility and expertise, and in terms of the impact of the proposed action on NOS activities and projects.

The following comments are offered for your consideration.

Section 2.3.1, Surface-water hydrology, has been found to be very adequate for the purposes intended. The authors are to be commended for the thorough bibliography on the subject.

NOS concurs with and encourages the oceanographic monitoring program described in Section 6.2.2. We feel this program will ensure environmental protection and help allay public concern.

A-8

U.S. DEPARTMENT OF COMMERCE
National Oceanic and Atmospheric Administration
National Marine Fisheries Service
Southwest Region
300 South Ferry Street
Terminal Island, California 90731

Date : January 8, 1979 FSW33/JJS

To : EC, Office of Ecology and Conservation

Thru : F7, Kenneth R. Roberts, Acting Director, Office of Habitat Protection

From : FSW, Gerald V. Howard, Regional Director, Southwest Region

Subject: Review of DEIS No. 7812.06 - San Onofre Nuclear Generating Station, Units 2 and 3 (NRC)

The subject DEIS which accompanied your memorandum of December 8, 1978, has been reviewed by the National Marine Fisheries Service. The following comments are offered for your consideration:

Specific Comments

5. Environmental Effects of Station Operation

5.4 Environmental Impacts

5.4.2 Impacts on the aquatic environment

Page 5-26, paragraph 7, Kelp

In paragraph 7 little information is included documenting the importance of kelp to coastal commercial fish species. Information available in the California Department of Fish and Game Fish Bulletin 139 (Quast, 1968) provides some insight in that regard.

Data developed by Jay Quast of the then U.S. Bureau of Commercial Fisheries, and included in that publication, indicate that in his studies he found more than twenty commercially important fish species occurring in the kelp beds off southern California. According to those studies the relationship of many of those species to the kelp habitat was more extensive than indicated by the final sentence of the subject paragraph. This should be reflected in the text of the final EIS.

6. Environmental Monitoring

6.3 Operational Monitoring Programs

6.3.3 Aquatic biological monitoring

Page 6-7, paragraph 1

The concept of continuing a kelp study program into the operational stage of SONGS is a good one. However, should those studies determine that significant harm is occurring to that resource, then some method of compensation satisfactory to the National Marine Fisheries Service would need to be developed. This should also apply to the studies being conducted on fish impingement at SONGS 2 and 3.

If such measures are not adopted and adverse impacts do appear the monitoring program may be merely documenting the demise of a valuable coastal resource.

10. Benefit-Cost Summary

10.7 Summary of Benefit-Cost

Page 10-3, paragraph 3

The potential additional cost of compensating for loss of biological resources due to the operation of SONGS 2 and 3 should be addressed.

LITERATURE CITED

Quast, Jay C. 1968. 8. Observations on the food of the kelp-bed fishes. In: California Department of Fish and Game, Fish Bulletin 139. Pp 109-142.

DEPARTMENT OF HEALTH, EDUCATION, AND WELFARE
PUBLIC HEALTH SERVICE
FOOD AND DRUG ADMINISTRATION
ROCKVILLE, MARYLAND 20857

January 25, 1979

Mr. William H. Regan, Jr., Chief
Environmental Projects Branch 2
Division of Site Safety
and Environmental Analysis
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Mr. Regan:

The Department of Health, Education, and Welfare has reviewed the health aspects of Draft Environmental Statement related to operation of the San Onofre Generating Station, Units 2 and 3, Southern California Edison Company and San Diego Gas and Electric Company and has the following comments to offer.

A-10

Section 5.5 Radiological Impacts

The individual doses as identified in Table 5.3 are all within Appendix I, 10CFR50 design objectives and should assure adequate radiation protection of the public for routine releases. However, it should be recognized in this section that 10CFR190 promulgated by EPA became effective in January 1979. A statement should be included indicating that SONG 2 and 3 will also meet this standard.

It is recognized that there are many variables that influence occupational exposure for a specific plant. However, a recent Atomic Industrial Forum study of occupational exposures showed for a PWR a total of 694 man-rem per year as a representative PWR exposure pattern. As a conservative estimate the projected occupational exposure impact of the two-unit San Onofre Station would be 1400 man-rem per year.

The summary of environmental consideration for uranium fuel cycle shown in Table 5.8 appear to be within acceptable radiation protection limits. However, some additional explanation within the test or by a footnote is needed for the disposal of solids. In particular, the statement that TRU and HLW would be buried at a Federal Repository should be modified to indicate alternatives for disposal of these waste in the event the Federal repository is not operational when required by plant operations.

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Page -2- Mr. William Regan

Section 6.2.5 Radiological Monitoring Program

The preoperational monitoring program presented in Section 6.1.5 of the Environmental Report is adequate for meeting the objective stated in this section. The establishment of the radiological monitoring program prior to start of operations should provide the necessary data to verify the effectiveness of in-plant controls and to provide assurance that undetected radioactivity will not build-up in the environment.

Section 7 Environmental Impact of Postulated Accidents

The estimated exposure of the population within a 50 mile radius of the plant shown in Table 7.2 cannot be adequately evaluated without some specific data in the text on the source term. Without such data it is possible to assume that the environmental consequence as a result of a class 8 accident could be more severe than indicated in the unlikely event of such an accident.

There is no indication in this section or previous chapters on emergency response planning to mitigate the consequences of an accident that could impact on the offsite population. A discussion of the arrangement that has been made with State and local authorities should be included in this section.

The discussions in paragraph 4, page 7-2 on the Reactor Safety Study (WASH-1400) relative to discussion with EPA is outdated (1973), and since it discusses scope of the study, and not results, should be removed. More importantly, a statement should be included on the technical review or conclusions that have been provided by EPA, other Federal agencies or independent reviewers. Such a statement would be helpful in accepting the low environmental risks associated with the postulated radiological accidents.

On the basis of this review it is concluded that the San Onofre Nuclear Generating Station, Units 2 and 3 can be operated to meet current radiation protection guidance and provide adequate protection of the public health and safety.

Sincerely yours,

Bernard Shleien for

Charles L. Weaver
Consultant
Bureau of Radiological Health

Southern California Edison Company
P. O. BOX 800
2244 WALNUT GROVE AVENUE
ROSEMEAD, CALIFORNIA 91770

J. H. DRAKE
VICE PRESIDENT

TELEPHONE
213-572-2222

February 2, 1979

Director, Office of Nuclear Reactor Regulation
Attention: William H. Regan, Jr., Chief
Environmental Projects Branch 2
Division of Site Safety and Environmental Analysis
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Gentlemen:

Subject: San Onofre Nuclear Generating Station
Units 2 and 3
Docket Nos. 50-361 and 50-362

A-11

In accordance with your request of November 30, 1978, the Southern California Edison Company and the San Diego Gas & Electric Company have reviewed the Draft Environmental Statement (DES) related to the operation of San Onofre Nuclear Generating Station, Units 2 and 3. Enclosed are comments generated from this review.

Should have any questions or require clarification regarding these comments, please contact me.

Very truly yours,

J H Drake

Enclosure

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ATTACHMENTS

A	Figure 6.14, page 61 of Reference (5)
B	Figure 6.29, page 76 of Reference (5)
C	Figure 6.34, page 81 of Reference (5)
D	Figure A-7, page A-15 of Reference (5)
E	Figure 6.8, page 47 of Reference (5)
F	NOAA Climatological Data, July 1975
G	" " , July 1976
H	" " , April and July 1977
I	" " , July 1978
J	Air Temperatures at SONGS
K	Del Mar Current Meter and Temperature Data, May-December 1978
L	San Onofre Area Current and Temperature Data, May-August 1978
M	Pages 103-106 of DES reference 8
N	Figure 1 Surface Isotherms for 0.0 knots
O	Figure 2 " " " 0.1 "
P	Figure 3 " " " 0.25 "
Q	Reference (19)
R	Reference (20)
S	Reference (21)
T	Temperature Data from References (8), (22), (23) and (24)
U	Reference (2)
V	Bottom (30') Water Temperatures at San Onofre, July and Aug. 1976-78
W	Pages 41 and 71 of Reference (16) and page 42 of Reference (17)
X	Revised DES Table 8.1
Y	Revised DES Table 8.3
Z	Revised DES Figure 3.5

AA	Reference	(1)
BB	"	(12)
CC	"	(13)
DD	"	(14)
EE	"	(18)

COMMENTS

BY

SOUTHERN CALIFORNIA EDISON COMPANY

SAN DIEGO GAS & ELECTRIC COMPANY

ON THE

DRAFT ENVIRONMENTAL STATEMENT

RELATED TO THE OPERATION OF

SAN ONOFRE NUCLEAR GENERATING STATION, UNITS 2 AND 3

DOCKET NOS. 50-361 AND 50-362

PREPARED BY THE

U. S. NUCLEAR REGULATORY COMMISSION

OFFICE OF NUCLEAR REACTOR REGULATION

TABLE OF CONTENTS

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A-14

INTRODUCTION

The Draft Environmental Statement (DES) has been reviewed by the Southern California Edison Company and the San Diego Gas & Electric Company (hereinafter referred to as Applicants).

Comments resulting from this review are to identify inaccuracies in the data or discussion and provide clarification or correction. The comments follow the organization and numbering in the DES and should be read in conjunction with the referenced section.

SUMMARY AND CONCLUSIONS

Comment A-1

(page iii, item 2)

The DES states, "Each unit will produce up to 3410 MWe and a net electrical output of 1057 MWe".

It should be noted that 1057 MWe as stated in the applicants' ER-OLS* and in the DES was calculated using the Turbine-Generator (T-G) manufacturer's guaranteed output of 1127 MWe, which corresponds to an NSSS output of 3266 MWe, and an estimated in-plant consumption of 70 MWe.

However, when the NSSS is operating at 3410 MWe, and the T-G is at the wide-open valve condition (normal operating condition) the T-G output will be 1181 MWe. Current estimates of in-plant consumption have been revised to 75 MWe. Therefore, it is suggested that the net electrical output value be expressed as being in the range of 1052 MWe to 1106 MWe per unit when the NSSS is operating in the 3266 MWe to 3410 MWe range respectively.

*ER-OLS will be revised to reflect the range of 1052 MWe to 1106 MWe output per unit, in a future amendment.

Comment A-2

(page iii, item 3a)

The applicants do not agree with the conclusion reached by the staff on the possible destruction of at least a portion of the San Onofre kelp bed as a result of the thermal discharge from San Onofre Nuclear Generating Station. The assessment of impacts to the aquatic environment is invalid because of the use of inappropriate data from the staff's numerical model. A reassessment of the impacts is needed using ambient temperatures from actual field data. Actual field data are appended to these comments. Using appropriate ambient temperatures in the assessment, the excess temperature from thermal plume predictions made by either the applicants or staff will not create adverse effects on the San Onofre kelp bed. (Attachment T)

Comment A-3

(page iv, item 6(B)(2))

The preoperational monitoring program outlined in Section 6 goes beyond the applicants' program approved by the NRC by letter dated July 6, 1978, and is apparently based on inappropriate predictions of impact to the San Onofre kelp bed. The operational monitoring program outlined in Section 6 is an extension of the preoperational monitoring program. The operational environmental monitoring programs are under development for Units 2 and 3 Environmental Technical Specifications (ETS) and will be submitted in the near future.

1. INTRODUCTION

1.1 HISTORY

Comment 1-1

(page 1-1, paragraph 2)

The net electrical output for each unit is in the range of 1052 to 1106 MWe. Refer to Comment A-1 for discussion.

A-16

2. THE SITE

2.4 METEOROLOGY

2.4.4 Atmospheric dispersion

Comment 2-1

(page 2-8)

The DES indicates that the onshore tracer test conducted by SCE substantiates the acceptability of data measured on the San Onofre onsite (bluff) tower for use in calculating atmospheric dispersion. However, the DES does not consider the enhanced dispersion estimates derived from the onshore tracer test results. Consideration should be given to the enhanced dispersion estimates derived from the onshore tracer test results.

2.5 SITE ECOLOGY

2.5.2 Aquatic ecology

Comment 2-2

(page 2-9)

Oceanographic data reports from the past have incorrect consultant sources referenced. The first source in the list of three sources should be:

"(1) a thermal effects study performed jointly by Environmental Quality Analysts, Inc. and Marine Biological Consultants, Inc. in 1973 using data and results obtained from 1964-1972 by Bendix Marine Advisors, Inc., and Intersea Research Corporation."

3. THE PLANT

3.2 DESIGN AND OTHER SIGNIFICANT CHANGES

3.2.1 Plant water use

Comment 3-1

(page 3-1, paragraph 2)

Delete the words, "makeup to," in the second sentence.

Comment 3-2

(page 3-1, paragraph 3)

The word, "makeup" should be corrected to "cooling," in the second sentence.

Comment 3-3

(page 3-1, paragraphs 2 and 3)

In the discussion of plant water use, the flushing of traveling bars and screens is incorrectly described. Seawater will be used for the flushing of the traveling bars and screens, not fresh water.

3.2.2 Heat dissipation system

Comment 3-4

(page 3-1)

The discussion should also include a description of the Seismic Category I Auxiliary Intake Structure of the circulating water system. The description of this design change can be found in Section 3.4.1 of the ER-OLS and Section 9.2.5 of the FSAR.

A-17

Comment 3-5

(page 3-1, paragraphs 4 and 5)

The seawater used for "cooling" has been incorrectly labeled "makeup." This error appears in the second sentence of paragraph 4 and the first sentence of paragraph 5.

Comment 3-6

(page 3-1, paragraphs 5 and 6)

The word "screenwell" should refer to the intake screenwell structure shown in Figure 3.3 and not the traveling screens. Lines 6 and 7 of paragraph 5 use "screenwells" where "traveling bars and screens" are being described. Also, in the second sentence of paragraph 6 "screenwells" is used instead of "traveling screens" and should be corrected.

Comment 3-7

(page 3-3, Fig. 3.2)

Figure 3.2 has been revised by the applicants to include the design details of the Auxiliary Intake Structure (Comment 3-4) and show the elimination of the manhole on the velocity cap. The revised figure can be found in Section 3.4 of the ER-OLS, Figure 3.4-2.

Comment 3-8

(page 3-5, paragraph 1)

The seawater used for "cooling" has been incorrectly labeled "makeup." This error occurs on lines 2 and 5, and should be corrected.

Comment 3-9

(page 3-5, paragraph 1)

The third sentence should read:
"To achieve the temperature required to control biofouling each unit has a recirculation and crossover gate."

2.3.1 Liquid Radioactive Waste Treatment System

Comment 3-10

(page 3-7, paragraph 7)

The flashed steam is routed to the "third point heater", not the "main condenser hotwell".

Comment 3-11

(page 3-8, Fig. 3.5)

The figure should be changed to reflect the correction identified in Comment 3-10. See revised Fig. 3.5 (Attachment Z).

Comment 3-12

(page 3-11, line 1)

The discussion on steam generator blowdown is incorrect. There are two steam generators for each unit, not four.

Comment 3-13

(page 3-13)

The discussion on the containment ventilation system should include a description of the 2,000 cfm mini-purge system. The description of this design change can be found in Section 9.4.1 of the FSAR.

2.4.1 Chemical Effluents

Comment 3-14

(page 3-16, paragraph 1, line 4)

The statement, "maintain a clean circulating water system," should be changed to read, "maintain a clean condenser system."

Comment 3-15

(page 3-16, paragraph 3, line 11)

The applicants will use a nitrite base compound or potassium chromate (K_2CrO_4) as the corrosion inhibitor for the turbine and component cooling water system. The statement in line 11, "will be treated with Nalco 39 to inhibit corrosion," should be changed to read, "will be treated with a nitrite based compound or potassium chromate to inhibit corrosion."

3.2.5 Transmission Lines

3.2.5.1 SCE Transmission Lines

Comment 3-16

(page 3-19, line 3)

The reference number for the description of retrofitting should be "4" not "1."

Comment 3-17

(page 3-20, Fig. 3.9)

An additional note should be added to the figure as follows:

> "The drawing depicts the four-circuit structure that will be used by SCE. SDG&E will use a similar structure with five circuits."

3.2.5.2 SDG&E Transmission Lines

Comment 3-18

(page 3-20, paragraph 1)

In the discussion of SDG&E's transmission lines, Talega Substation has been misspelled consistently throughout.

3.2.6 Probable Maximum Flood Berm

Comment 3-19

(page 3-23, line 1)

The reference number for the letter to the NRC should be "5" not "4."

5. ENVIRONMENTAL EFFECTS OF STATION OPERATION

5.3 IMPACTS ON WATER USE

5.3.1 Thermal discharges

General Comment Concerning Section 5.3

Applicants and the NRC both have evaluated the thermal effects of the diffuser system proposed for SONGS 2&3. The applicants applied a physical hydraulic model study. The NRC staff applied a depth-averaged numerical model. Applicants' model predicts compliance with all state and federal water quality requirements. The NRC Staff model predicts similar compliance for all realistic conditions but predicts potential violations of state thermal regulations for certain admittedly unrealistic conditions.

For reasons inherent in the input and methodology of the NRC staff model, applicants do not consider the staff's predictions to be valid. Further, applicants' model does not predict violations of the State Thermal Plan even under the unrealistic conditions postulated by NRC staff. Specific comments on DES Section 5.3 are discussed below:

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5.3.1.1 Applicant's thermal analysis

Comment 5-1

(page 5-1, paragraph 6)

In the discussion of the physical model, the temperature difference of the discharged water is reported to be 30°F higher than the surrounding water. The 30°F delta T was necessary to achieve dynamically correct scaling of the actual delta T of 20°F and this fact should be mentioned in the discussion to avoid confusion.

Comment 5-2

(page 5-2, paragraph 3)

The statements are made, "The staff has reviewed the applicant's thermal analysis and believes that the physical model does not adequately represent certain hydrodynamic mechanisms and certain physical features of the prototype. The most significant of these is the limitation of the duration of the simulation by the size of the model basin." and "In fact, for the conditions represented in Figure 5.3,

an increase in simulation time would likely have resulted in predicted excess temperatures that violate state thermal standards." The applicants do not agree with these statements. The assumption that the size of the model basin limits the ability of the model in terms of representing valid results for longer time duration conditions are not considered to be valid. The conditions represented in DES Figure 5.3 represent a worst case condition and it is illustrated in the following paragraph that equilibrium had already been reached. An increase in simulation time would not have changed the predicted results.

In Figure 6.14, page 61 of Reference (5) (Attachment A) it is shown that for the circumstances represented in the DES Fig. 5.3, the hydraulic model had clearly reached an equilibrium peak temperature. The prototype period of time represented in this hydraulic model test of a zero drift situation is in excess of 30 hours of continuous operation at full load. Referring to Attachment A it can be seen that the peak temperature measured in the hydraulic model basin (the upper curve) quickly reaches an equilibrium level in a time of approximately 12 prototype hours. For the subsequent 18 hours of operation at zero drift velocity, the only variation in temperature is that associated with the experimental fluctuations. The behavior is similar for the lower curve, which is the peak temperature measured at a distance equivalent to anywhere beyond 1000 ft. from the point of discharge.

The results given in Attachment A, and the detailed error analysis performed in Reference (5), show quite clearly that there is no basis for the assertion that the hydraulic simulation represented in Fig. 5.3 of the DES, if continued, would lead to a violation of the state thermal standards. To the contrary, it is clear that in a no-drift situation an equilibrium peak temperature of 2.3°F (beyond 1000 ft.) would be reached within about 12 hours and this peak temperature would not be exceeded for longer durations.

The DES further states, "Once the thermal plume reaches a lateral boundary of the tank, the simulation must be terminated. The length of the simulation is thus dictated by the size of the model basin rather than by the natural time scales of the problem."

The tests do not have to stop when the thermal plume reaches a boundary because a large prototype area is represented by the model basin and the maximum temperatures are close to the diffusers. Furthermore, the natural time scale of the problem is that associated with the initial jet mixing and establishment of the steady induced offshore drift of the

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thermal field. The time scale associated with the establishment of steady state conditions in the model was found to be 12 hours at the most. The size of the basin does not limit the results until more than 40 hours, as shown in Attachment A. It is also confirmed by the results given in Figures 6.29 and 6.34, pages 76 and 81 of Reference (5) (Attachments B and C). The results shown are for a situation where the hydraulic model was operated for the accelerating current pattern given on Attachment B. The outcome of the model is shown in Attachment C. It can be seen that the peak temperature rapidly reduces as the current velocity increases, showing that the natural time scale or response time is only a few hours. Indeed, it is because of the short time scales of the problem that the hydraulic model is appropriate.

The reason for the short time scale can be seen in Figure A-7, page A-15, of Reference (5) (Attachment D) and in Figure 6.8, page 47, of Reference (5) (Attachment E) which both clearly show a surface layer of warm water overlying a cooler bottom layer. The diluting water for the discharge jets is always drawn from this cooler bottom layer so the dilution is fixed by the rate of supply of bottom water. When there is no drift the bottom flow is generated by the jet entrainment. When an ambient current is present the flow of diluting water is even greater so the peak temperatures are reduced.

Comment 5-3

(page 5-2, paragraph 4)

The DES states, "Although the problem of underprediction is inherent in all the applicant's results, it is less significant for the realistic cases." It cannot be concluded that the hydraulic model consistently underpredicts delta T's with respect to what will really occur; rather, the only conclusion that can be drawn is that the math model gives consistently higher values than the physical model.

The basic hydraulic model report (Reference (5)) discusses possible errors in hydraulic modeling and deduces a laboratory target value of 2.5°F so that all possible errors will be included within the 4°F limit; but the report does not imply that there is an expected bias in the results as the errors could as well be negative as positive.

5.3.1.2 Staff's thermal analysis

Comment 5-4

(pages 5-3, 5-4, and Fig. 5.6)

Atmospheric data purported to be typical of July are used in the NRC mathematical model to predict ocean temperatures. Specifically, air temperatures with a maximum of approximately 82°F and a minimum of 65°F were used in the model (see DES Fig. 5.6, page 5-7).

Actual field data measured at coastal sites in Southern California for July show mean daily maxima and mean daily minima substantially lower than these temperatures. In addition, temperature summaries for the San Onofre site presented in Table 2.3-6 of the FSAR and Table 2.3.2-4 of the Applicants' Environmental Report OL Stage show mean daily maxima and mean daily minima temperatures on the order of 67°F and 61°F respectively.

Published U.S. Climatological Data for July 1975, 1976, 1977, 1978 (Attachments F, G, H and I) give temperatures for two coastal stations (Newport Beach Harbor and Santa Monica Pier). Data at San Onofre are from the meteorological tower maintained at the site: (Attachment J).

Actual Air Temperatures For The Month Of July

	Newport Beach Harbor mean daily		Santa Monica Pier mean daily		San Onofre mean daily	
	max	min	max	min	max	min
1975	69.8°F	62.1°F	68.0°F	61.5°F	66.6°F	59.5°F
1976	72.4°F	64.4°F	71.1°F	63.5°F	67.6°F	63.3°F
1977	70.7°F	61.4°F	67.9°F	61.2°F	67.5 F	61.5°F
1978	70.7°F	62.0°F	68.1°F	59.8°F	67.5 F	61.2°F

The indication is clear that a typical July daily atmospheric maximum temperature at San Onofre should be in the order of 68°F with a typical minimum of about 61°F.

These atmospheric data are an important feature of the numerical model since high air temperatures will lead to high ambient water temperatures being produced by the numerical model in the inshore region. An indication that this has in fact occurred, are the water temperatures used in Section 5.4 (in the benthic section ambient depth-averaged temperatures of 27.8°C (82°F) and in the kelp section ambient bottom temperatures of 21.5-24°C (71-75°F)). These temperatures are considerably higher than have actually been measured in the field (References (8),(22) and (23)). High ambient water temperatures in the inshore region will, in turn, be reflected as high temperature increments offshore due to the inshore water being transported offshore by the net offshore drift produced by the diffusers. It is quite likely that these features of the numerical model could be responsible for the possible temperature excess violations predicted by the staff's numerical modeling.

<u>Comment 5-5</u>

(page 5-4, paragraph 2)

The DES omits computed ambient temperature maps (without heated water discharge) and computed temperature maps with thermal discharge from which the delta T maps were derived as presented in DES Figures 5.8, 5.10, 5.12, 5.14, 5.16, 5.18, 5.20, 5.22. DES Section 5.4, environmental impacts, refers to this section (5.3.1) and discusses absolute values of ambient temperatures.

Since the basis for the prediction of temperature excess associated with the operation of SONGS Units 2 and 3 is the difference between the numerically predicted temperature distributions for operating and ambient conditions, these temperature maps should be made available to the applicants for evaluation and interpretation, or included in the FES. In addition, these temperature maps are essential to the assessment of impacts on marine life and necessary to provide the basis for much of DES Section 5.4.

<u>Comment 5-6</u>

(page 5-7, paragraph 2)

The DES states, "The net downcoast drift used for these simulations is based on limited data for mid-July. During other times of the year, the data indicate that an absense of drift can persist for up to several days. Although there are no data to confirm a no-drift assumption during mid-July, the staff believes that this situation is at least possible, and therefore, should be considered." Applicants disagree with the assumpation that a no-drift situation is possible.

Current data analyses have been previously supplied to NRC (References (3) and (4)). In Reference (3), pages 59 and 60, it was concluded that current speeds are higher in summer than in winter and that, during winter, periods of very low currents could exist lasting a few days, but that tracks indicated no evidence of currents with no net transport during this period. The available current record for summer, published in Reference (3), shows no evidence of any period of no net drift.

In Reference (4) more recent data obtained by Winant and Severance for the Marine Review Committee were analyzed. These data were collected using a newer type of current meter less susceptible to clogging than the meters used for the data previously analyzed (obtained from Intersea Research Corporation). Reference (4) makes it clear that at no time in the current meter records are there data to indicate that there is a period of zero drift. In fact, the records indicate a substantial drift either upcoast or downcoast with a speed of the order of 0.1 to 0.2 knots (5-10 cms/sec). The data therefore confirm the drogue and current meter data initially obtained by Intersea Research Corporation (IRC). The data appear to indicate that in fact IRC's meters may have been underrecording the magnitude of the currents.

In the past year (1978) more data have been collected at Del Mar (15 miles downcoast from San Onofre) by Winant of Scripps Institution of Oceanography and also at San Onofre by Brown and Caldwell Engineers under contract to the Marine Review Committee of the California Coastal Commission. Winant's data (Attachment K) show substantial longshore currents always occur at Del Mar. The Brown and Caldwell data obtained at the San Onofre site appear to be well correlated with the Del Mar data and also indicate strong drift currents either upcoast or downcoast for periods of several days. The change in direction is always a rapid process. These most recent data further corroborate the previous conclusion that there exist no periods of zero drift (Attachment L). A zero drift period is not considered to be credible, and should not be postulated for evaluating compliance with the state thermal requirements.

Comment 5-7

(page 5-7, paragraph 3)

The DES states, "Although the thermal numerical model is depth-averaged, it is still possible to address the state standards with model results because the buoyancy and shear generated by the diffuser discharge produce a hydrodynamic instability, resulting in the water column's being well mixed within several diffuser lengths of the discharge[8]. Therefore, within the specified mixing zone, the depth-averaged predictions are reasonable representations of surface temperatures."

Reference 8. C. W. Almquist and K. D. Stolzenbach, Staged Diffusers in Shallow Water, Report No. 213, Ralph M. Parsons Laboratory for Water Resources and Hydrodynamics, Massachusetts Institute of Technology, Cambridge, Massachusetts, 1976.

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Referring to pages 103-106 of DES Reference 8 (Attachment M) it is clear that the hydrodynamic instability claimed as the basis for application of depth-averaged numerical modeling definitely will not occur with the San Onofre diffusers. It is therefore evident that depth-averaged modeling is inappropriate to the San Onofre configuration so that drawing conclusions about violation of the California thermal standards on the basis of the results of such a model is not valid. It cannot be concluded that depth-averaged predictions are reasonable representations of surface temperatures. For the same reasons, the bottom temperatures cannot be predicted correctly from the NRC depth-averaged numerical model.

Comment 5-8

(page 5-7, paragraph 4)

The DES states that, "The model numerical is inadequate for addressing the issue of bottom temperature. However, at worst, the 4°F excess temperature should only touch the bottom over a very limited area in the vicinity of the Unit 2 and 3 diffusers."

The applicants agree that the numerical model is inadequate for addressing the bottom temperature issue as noted above. In view of the staff's admission of this inadequacy, there is no basis for the staff's statement concerning the 4°F excess bottom temperatures. In the assessment of San Onofre 2&3 diffuser plume extent, Figures 1, 2 and 3 have been formulated from Reference (5) (Attachments N, O and P). These show hydraulic modeling results in the horizontal and vertical and with respect to the San Onofre kelp bed area under conditions of no ambient currents, and two typically encountered downcoast ambient currents.

It should be noted that the vertical profiles in Figures 1, 2 and 3 (Attachments N, O and P) stop at a depth of 35 feet but the actual bottom depth is deeper. These figures show no indication of impingment of a 4°F isotherm on the bottom or even present in the water column.

Figure A-7 (Attachment D) shows an actual vertical cross-section of the modeling results from surface to bottom along the centerline of the San Onofre 2 & 3 diffusers. This figure shows that the thermal plume does not impinge substantially anywhere on the bottom and that a temperature increase of 0.4°C (0.8°F) is not exceeded on the bottom.

Comment 5-9

(pages 5-2 through 5-24)

The DES omits any reference to error analysis for either applicants' hydraulic modeling or the staff's numerical modeling. Such an analysis is essential in determining the bounds within which the results are accurate or applicable.

The applicants' modeling has been subjected to a comprehensive error analysis, Reference (7), which discussed possible sources of error and determined appropriate error margins. This error analysis should be referenced in the DES.

There is no discussion of errors for the staff's math model. As with all math models, various assumptions and coefficients are necessary and the results must be viewed with consideration of the potential error inherent in the model. It is a particular concern with this math model which appears to be deficient in representing the real phenomenon occurring, specifically two-layer stratified shear flow and individual jet mixing. Because of this, the range of possible error for the math model is considered to be greater than for the hydraulic model. An error analysis for the staff's math model should be presented, or at least referenced and made available to the applicants.

Comment 5-10

(page 5-7, paragraph 5)
(page 5-24, paragraph 1)

The DES states, "In the absence of drift, the 4°F excess temperature will not reach the shore. However, state thermal standards would be violated since the 4°F surface isotherm will extend beyond the 1000 ft. radius during most of the tidal cycle. The staff concludes that although there exists a remote possibility that state thermal standards could be violated by the operation of Units 2 and 3, violations would, at worst, be infrequent and for short periods. There is no evidence in available drift data to indicate that such an occurrence would take place during the summer when thermal impacts would be the most severe."

The applicants do not agree that the state thermal standards limitation for the 4°F surface isotherm beyond 1000 ft. for more than one half of a tidal cycle will be violated in the absence of drift or under any other circumstance. The applicants' thermal modeling studies addressed a no-drift condition, showing no violation of state thermal standards (DES Figure 5.3). It is the position of the applicants that the mathematical model predictions are excessively high, mainly as a result of inappropriate inputs and assumptions. The staff selected inputs include air temperatures that are about 10°F too high (see Comment 5-4), unsubstantiated two day no-drift conditions along the open coast at San Onofre in July (see Comment 5-6), and modeled ambient depth average water temperatures that are higher than ever recorded in the area's field data (see Comment 5-8). Also, such violations predicted (as remote) by the staff are derived from output of their mathematical model when the model itself could be approaching its limits of validity. Yet, this can not be proven, mainly because an error analysis that would substantiate the claimed applicability of the numerical model is not included in the DES (see Comment 5-9).

For these reasons, the staff's conclusion, that even a remote possibility of a violation of the state thermal standard exists, cannot be justified on a technical basis.

5.4 ENVIRONMENTAL IMPACTS

5.4.1 Terrestrial environment

Comment 5-11

(page 5-24)

The discussion on environmental impacts to the terrestrial environment should also include an assessment of the Probable Maximum Flood (PMF) berm. The applicants submitted an environmental assessment of the PMF berm, by letter dated February 14, 1978. The assessment indicated that the PMF berm should have no adverse environmental impact on the terrestrial ecological characteristics.

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5.4.2 Impacts on the aquatic environment

5.4.2.1 Effects of the heat dissipation system

Thermal effects

Comment 5-12

(page 5-24, paragraph 8)

The discussion of the proposed heat treatment states, "the applicant proposes to heat treat portions of the intake system to remove biological growth (Sect. 3.2.2)." This statement is incomplete since the applicants also propose to heat treat the discharge system. The text should be changed to reflect this point.

Comment 5-13

(page 5-25, paragraph 2)

While the applicants do not agree that the area to be affected by thermal discharges will be greater than previously thought, the applicants do concur that even assuming a larger plume, the impact to the aquatic environment is expected to be minimal and of an acceptable nature.

Fish

Comment 5-14

(page 5-25, paragraph 5)

The applicants agree that cold kills of fish are not likely to occur, but for the reason that the extent of the thermal plume is relatively small, and the difference between the ambient and the induced temperatures is less than the rapid temperature changes that occur naturally.

Comment 5-15

(page 5-25, paragraph 4)

It is stated that, "However, with more area to be influenced by the effluent, more fish potentially will be affected."

This appears to be an oversimplification since the thermal plume will not be uniformly distributed with depth but rather the more buoyant heated water will be on the surface with the bottom water remaining unaffected. This means that an increase in the surface area of the plume would only effect fish species which inhabit the upper water column and no additional effect would be expected for fish associated with the bottom. Fish are not uniformly distributed within the water column and actually exhibit a distribution opposite to that of the thermal plume, that is with a greater concentration of fish associated with the bottom and fewer fish associated with the surface. During the 1976 ETS studies a comparison of surface versus bottom gill net data showed that 88% of the fish were found on the bottom and only 12% in the surface waters. Therefore, the area potentially effected by a larger plume would be only the surface waters, which have a relatively small percentage of the total fish abundance.

Benthic fauna

Comment 5-16

(page 5-25, paragraph 8)
(page 5-26, paragraph 1)

In the discussion of DES reference 11 (Ford, et al., 1976), it is not made clear that effects upon growth and mortality of *S. franciscanus*, *P. ochraceus* and *R. poulsoni* occurred only in the experiment simulating a location 84 meters from the discharge and not at 335 m away.

The applicants recommend that clarification be added to these paragraphs in order that the reader be clearly informed that the effects discussed in the DES were limited to the simulation of the 84 location meter distant from the point of discharge.

Benthic fauna

Comment 5-17

(page 5-26, paragraph 2)

Ambient water temperatures in DES Section 5.3.1 are referenced here but no ambient temperatures are included in that section. As previously noted, in Comment 5-5, maps of these ambient temperatures should be presented.

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The ambient temperatures used in the discussion of the assessment of benthic fauna are apparently taken from the staff's mathematical model. The ambient temperatures used are clearly too high, as example, "temperatures potentially as high as 27.8°C (82°F) may occur naturally,..." This is far in excess of actual measurement of natural ambient water for the area.

The maximum surface water temperature reported in the vicinity of San Onofre is approximately 23°C (References (8),(22), (23), (16) and (17)). Mean San Onofre natural surface temperatures for July and August of the past three years are on the order of 19°C and the corresponding bottom temperatures are about 17°C.

University of California Scripps Institution of Oceanography (SIO) data reports entitled "Surface Water Temperatures at Shore Stations - United States West Coast" give mean surface water temperatures at San Clemente pier, five miles North of San Onofre, References (16) and (17)):

Mean Surface water Temperatures at San Clemente

	July	August	September
1977	18.27	20.48	18.53°C
1976	19.59	17.95	19.77
1975	18.58	17.01	17.91
3 year mean	18.8	18.5	18.7°C

With surface temperatures in the 18-19°C range it should further be noted (for benthic assessment) that corresponding bottom temperatures will be even lower: all San Onofre field data support the existence of vertical temperature stratification in depths greater than about 30 feet when surface temperatures are in this range. (see Attachment T).

Comment 5-18

(page 5-26, paragraph 3)

The DES states, "On the basis of the 1977 study[11] the staff concludes that several components of the benthic fauna in the vicinity of SONGS would probably be adversely affected in areas where weekly mean temperatures of 22°C prevail for one month or more or where daily temperatures reach or exceed 24°C. It is not, however, anticipated that temperatures averaging 22°C will occur for more than 2 to 3 weeks or that the areas experiencing temperatures of 24°C or greater as *a result of SONGS operation will be considerably larger than* the area experiencing these temperatures under natural conditions."

Based upon historical temperature records between 1975 and 1978 (References (8),(22) and (23)) the use of weekly mean bottom temperatures of 22°C appears to be inappropriate and should be lowered to 17°C.

The applicants recommend, therefore, that the sentence indicating 22°C weekly mean temperature could exist on the bottom for 2 to 3 weeks be changed and that a summary sentence be added to indicate that the components of the benthic fauna previously discussed will not be adversely affected.

Also, the date of DES reference 11 (Ford, et. al.) is 1976, not 1977, as stated in the first sentence of the paragraph.

Kelp

Comment 5-19

(page 5-26, paragraph 5)

The DES states, "Although this deterioration may have been partially a result of overharvesting, much of it is probably caused by the increased alteration of the near-shore environment by human activities. In particular increased temperatures and increased turbidity have been shown to be inimical to kelp survival."

The thermal effects on kelp cited in Phillips (1974) were for naturally occurring events and not as induced by human activities. Man induced thermal effects on kelp have not been demonstrated.

The turbidity comment by Phillips (1974)(Reference (15)) was in reference to work conducted by North (1960)(Reference (12)) on effects of sewage outfalls on kelp health. The type of *turbidity generated by a sewage outfall is not equivalent* to the surface turbidity which may be associated with a cooling water discharge.

It is recommended that the discussion be changed to reflect that the deterioration may have been partially a result of overharvesting, much of it is probably caused by increased alteration of the near-shore environment by human activities,

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in particular, sewage treatment facilities and industrial/chemical discharges. The toxic element of each discharge has not been isolated to date, i.e., heavy metals, sedimentation, oils, turbidity, etc.

Comment 5-20

(page 5-26, paragraph 6)

The DES states, "Typically, canopy tissue deteriorates during the warmest time of the year leaving the basal portion of the plant (which is in cooler water) for regeneration when temperature and light conditions permit."

It has been documented that kelp deterioration occasionally occurs when (apparently) surface temperatures exceed critical thermal limits. However, it appears that seasonal kelp deterioration may be due to synergistic effects and not just to a thermal component. In the open coast setting, an inverse relation often occurs between temperature and dissolved nutrients. As the temperature increases, the nutrient content often decreases, to or perhaps below levels critical to kelp. Additionally, the highest nutrient concentration is found on the bottom near the basal tissues and the lowest concentration near the surface where most kelp deterioration occurs (Reference (2)).

Other evidence (Reference (13)) implies that when Macrocystis pyrifera is placed in a bay, which are typically much higher in nutrients than found in the open coast, the kelp remains in the healthy state even when the entire plant is subjected to 25-26°C (77-79°F) for extended periods.

At this time, it is not known clearly if temperature, nutrients, and/or other unknown components of the water contribute the most to kelp deterioration. However, there is a possibility of a beneficial effect from Units 2 & 3 operation if outfall upwelling creates a surface nutrient plume that will occasionally come in contact with kelp plants during the warm water months.

It is recommended that the DES be changed to reflect the fact that typically, canopy tissue deteriorates during the warmest time of the year leaving the basal portion of the plant (which is in the cooler water) for regeneration when temperature and light conditions permit; and that reduced surface nutrients and higher bottom nutrient concentrations may contribute to canopy deterioration and basal tissue regeneration, respectively (Reference (2)).

Comment 5-21

(page 5-26, paragraph 7)

The DES states, "It is estimated that the larval, juvenile, and adult stages of 25 main sport fish use kelp beds for refuge and food gathering (eating the associated invertebrates, the kelp itself, or other algae) and the average standing crop of fish is estimated to be 300 kg/ha (300 pounds per acre)."

For many years it was believed that the kelp beds, especially the canopy region, represented spawning and nursery grounds of many sport and forage fish (Reference (1)). No evidence is available to support the therory that the canopy is widely used as a spawning area (Reference (6)). Larvae of a few fishes are found in greater abundance in kelp beds than elsewhere. These include the topsmelt, kelp goby, kelp clingfish and striped kelpfish (Reference (1)); species not considered important sport fish.

Many juvenile fishes inhabit the kelp canopy. However, those of recreational or commercial value are found to be more numerous in rocky areas away from kelp, i.e., kelp bass. The only common juvenile fish that are reported to be at higher concentrations within kelp beds are kelp surfperch, kelp pipefish and kelp clingfish (Reference (1)).

Only one adult species, the kelp clingfish, is considered to be obligate to kelp plants. All other fish species will persist in the environment with or without kelp plants present. Diversity of fish species is not altered significantly by the presence or absence of kelp. A highly varied bottom topography appears to be the most important factor for extensive fish-life and to be of greater significance in this respect than kelp (Reference (14)).

The DES should be changed to reflect the fact the kelp beds do not appear to be spawning grounds, rearing grounds, or refuge areas for recreationally or commercially important fish species (Reference (1) and (14)). Only the kelp clingfish appears to be obligate to kelp beds for survival.

Comments 5-22

(page 5-26, paragraph 8)

The DES states, "Kelp is an important commercial commodity ...harvested yearly at a landed value of $2 million."

The commercial value of kelp is well documented, although, a kelp bed is only considered commercially important when it has: high stand density, extensive areal coverage and close proximity to a commercial harbor. The San Onofre kelp bed does not now nor has it ever met these criteria because of the limited extent of substrate suitable for attachment. The DES should be revised to reflect the fact that the kelp beds in the vicinity of San Onofre are not commercially harvested.

Comment 5-23

(page 5-27, paragraph 2)

The DES states, "It has been rather well established that temperatures above 18-20°C (64-68°F) cause deterioration of kelp, and the degree of degradation is directly related to the duration of exposure to these temperatures. Increased surface temperatures caused by SONGS operation (all three units) would have the effect of extending the period of canopy absence. During the hottest time of the year, data in Section 5.3.1 suggests that the closest kelp bed (San Onofre bed) will experience an average surface temperature increase (over a 24-hour period) of 1.4°C (2.6°F); the range of temperature increase will be 0.6-2.2°C (1-4°F)."

The statement in Reference (15), of 18-20°C (64-68°F) thermal exposure causing kelp deterioration was based on comments made in Reference (12), which refers to the colder water variety of kelp found near Monterey, California. For kelp plants located in southern California waters, the critical thermal maximum is more in the range of 20-22°C (68-72°F) (Reference (21)).

During the warm water months of the year, data in Section 5.3.1 suggests that the closest kelp bed (San Onofre bed) will experience average surface water temperatures increases due to the operation of SONGS of less than 0.6°C (1°F); the range of temperature is 0-0.9°C (0-1.5°F).

Temperatures taken in the vicinity of San Onofre between July and September over a three year period show a range of averages of 18.5 to 18.8°C (65.3-65.8°F) for the surface waters (References (16) and (17)). Clearly, the predicted maximum temperature increase of 0.9°C from plant operation when added to the ambient temperature in the vicinity of San Onofre of 18.8 C will not exceed the critical thermal limits established by North. The DES should be revised to reflect this fact.

Comment 5-24

(page 5-27, paragraph 3)

The DES states, "Although daily natural temperature variations of 1°C (2°F) are not uncommon in the area (ER page 2.2-28) they would not be continuous in nature and would thus not affect the bed as severely as the continuous SONGS discharges would. Prediction of the degree to which canopy disappearance would be prolonged is impossible. Regeneration would be quicker in years with naturally cooler ocean temperatures, assuming the regenerative tissues remain unaffected (see below)."

The operation of SONGS 1, 2, and 3 will not have a continuous effect on the San Onofre kelp bed. Power plant thermal discharges will contribute no more than 0.9°C surface temperature increases to the kelp bed and thus will only occur with downcoast currents. The more recent current meter data, as discussed in Comment 5-6 must be considered in regard to this kelp section. It is seen from these data that summer upcoast currents, which would result in no kelp bed plume impingement, occur during approximately half of the summer season. Further, the increase in temperature will be dependent on the strength and duration of the current. Increased surface temperatures due to the operation of SONGS 1, 2 and 3 will always be less than the measured natural surface temperature variations of the area, and will be sporadic.

The staff is requested to revise the DES to reflect the fact that increased nutrients brought to the kelp bed surface waters by outfall induced upwelling may help resist the natural seasonal canopy deterioration and provide beneficial effects from station operation when an outfall induced nutrient plume drifts over the kelp bed during warm water months.

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Comments 5-25

(page 5-27, paragraph 4)

The DES shows ambient bottom temperatures in July reaching as high as 23-24°C (74-76°F) with temperature of 22-23°C (72°F and 73°F) for a week at a time. These temperatures are the outcome of the staff mathematical model (DES Section 5.3.1) and are an inaccurate representation of existing natural conditions occurring at San Onofre. Applicants' Comment 5-17 suggests that a bottom temperature of 17°C (63°F) is a more realistic representation.

Also, this section references DES Section 5.3.1 as a basis for a typical bottom temperature range of up to 19°C (66°F) in August and September, however, these referenced temperatures are not found in DES Section 5.3.1. Such a temperature appears to represent more adequately the extreme or high end of the range of summer bottom temperatures at San Onofre. As indicated above, an appropriate representation of a monthly or weekly mean bottom temperature would be 17°C (63°F).

Comment 5-26

(page 5-27, paragraph 4)

The DES states, "...a several week period could exist in which temperatures exceed 19°C."

Results of the applicants' thermal analysis demonstrates that the temperature increase at the bottom in the San Onofre kelp bed will be much less than 0.6°C (1°F) under any current condition. Under most conditions it is predicted that there will be no increase in bottom temperature in any portion of the kelp bed. Bottom temperatures measured at San Onofre during July and August over a three year period show a range of averages of 12-18°C (55-64°F). The addition of less than 0.6°C (1.0°F) to measured ambient temperatures should have no adverse effects to kelp basal tissues from which the canopy is regenerated annually.

Comment 5-27

(page 5-27, paragraph 5)

This paragraph summarizes the staff's conclusions that, based on assumed natural bottom temperatures of 21.5 - 24°C (71 - 75°F) and bottom temperature increases in the San Onofre kelp bed of 1 - 1.5°C (2 - 3°F) due to operation of Units 1, 2 & 3, damage to the kelp basal tissue might result if slack currents occur for several days. Further, if this scenerio occurs frequently, the bed might not recover fully, resulting in long term damage. While the staff admits this is unlikely, it recommends additional extensive monitoring of the San Onofre kelp bed.

It is the applicants' conclusion that an assessment based on appropriate ambient bottom temperatures (17°C or 63°F) derived from actual field data, and temperature increases recognizing that the thermal plume will be stratified (0.6°C/1.0°F maximum) will yield a conclusion that damage to basal tissues will not occur, even under worse case conditions. Also, there is no evidence to support the use of an assumption that a condition of several days of slack current will ever occur, or that it would occur frequently. The applicants believe that the proper conclusion to be drawn from the relevant data is that the operation of San Onofre Units 1, 2 & 3 will have no significant adverse effects on the San Onofre kelp bed.

The greatest adverse effect which could be expected is a slight prolongation of the natural summer surface canopy deterioration period which does not effect the basal tissues or the regeneration of the kelp in the fall.

Based on the above evaluation, the extensive monitoring recommended by the staff is not justified, and monitoring presently being accomplished is sufficient to assess potential effects of San Onofre Units 1, 2 & 3. Specific comments on the monitoring are contained in Comment 6-3.

A-29

5.4.2.1 Turbidity and Sediment Transport Effects

Comment 5-28

(page 5-27, paragraph 6)

The DES is deficient in that it fails to substantiate the assertion that larger thermal plumes directly imply larger turbid plumes.

Comment 5-29

(page 5-28, paragraph 1)

The DES states, "The effect on the kelp would potentially be decreased photosynthesis, possibly causing many of the plants to die if the exposure is continuous (a 1% increase in the absorption coefficient has been found to result in a 20% loss in net photosynthesis at 15m)[13] and burial of the holdfasts in particles which settle out, inhibiting regeneration and recolonization. Regardless of the magnitude of these effects, their presence would add to the probability that the kelp bed would be adversely affected (see preceding section)".

As discussed in Comment 5-24, the plume from SONGS 1, 2 and 3 will not have continuous contact with the San Onofre kelp bed.

Reductions in photosynthesis from power plant induced turbidity has not been demonstrated. The net reduction in photosynthesis referred to by Phillips (1974)(Reference (15)), was based on work by North (1958)(Reference (18)). The model (computation) was based on a uniform dispersal of light absorbtive material throughout the water column. This model was designed for the turbidity generated by a sewage outfall. For thermal diffusers, there would be an uneven distribution of natural turbidity and the equation does not apply.

Sewage outfalls generate a substantial amount of turbidity that is dispersed throughout the water column. A thermal outfall does not create turbidity, but rather, can occasionally redistribute portions of a naturally occurring dense bottom turbid layer to the surface. Therefore, there is no net gain in the amount of suspended matter in the water. The major effect is that the turbidity on such occasions can be seen on the surface. Further, the turbid plume characteristics sometimes experienced at Unit 1 should not be applied to Units 2 and 3.

A surface plume can be seen at Unit 1 when the surface waters are relatively clear and the bottom water is turbid. The intake and outfall withdraws and upwells, respectively, portions of the bottom turbid layer and pumps it to the surface. The bottom turbid layer qualitatively appears to be essentially a nearshore phenonemon that is generated from wave agitated bottom sediments. Units 2 and 3 outfalls are located in deeper and clearer ocean waters, although the intakes are at a depth comparable with Unit 1. It is predicted that on occasions when naturally occurring turbidity is present the Units 2 and 3 intakes will withdraw turbid bottom water like Unit 1, however, the Units 2 and 3 outfalls will be upwelling clearer bottom waters. Additionally, Units 2 and 3 effluent will be initially diffused through 63 ports each and then mixed with the receiving water at an estimated ratio of 10:1 (Unit 1 dilution ratio is approximately 3:1). Given the situation of clearer water at the outfalls and increased mixing of effluent, it is predicted that a turbid plume will not normally be detected.

In terms of effects, Unit 1 can be viewed as potentially creating more severe effects than Units 2 and 3, i.e., single port outfall and reduced mixing (3:1). The environmental evidence indicates that there is no adverse impact on benthic faunal or floral groups near the outfall. In fact, results from the Environmental Technical Specifications benthic program demonstrate that the fauna and flora near the Unit 1 outfall are more abundant than those from the control station (References (8), (9) and (10)).

No relationship has been established between a turbid plume and thermal plume. The factors that influence the intensity and extent of each constituent are different and may not be interrelated.

The applicants' conclusions are that a turbid plume emanating from Units 2 and 3 operation may occur under certain oceanographic conditions, however, it should be less intense than observed at Unit 1 because (1) of increased mixing of the discharge and (2) the diffusers are located in deeper, clearer waters. Environmental Technical Specifications benthos study results show that redistributing a natural turbid layer has no adverse effects on faunal and floral groups for Unit 1 (References (8), (9) and (10)). Therefore, no adverse effects on faunal or floral biota are predicted.

A-30

Entrainment

Comment 5-30

(page 5-29, paragraph 2)

The DES states, "The staff's analysis of entrainment effects in the FES-CP remains valid (FES-CP, p. 5-7 to 5-12). A program on the mortality experienced by entrained ichthyoplankton is being planned currently at SONGS 1 and is expected to be submitted to the NRC staff in December, 1978, for approval."

Refer to (applicants' Comment 6-5).

Impingement

Comment 5-31

(page 5-29, paragraph 4)

The DES states, "The majority of the fish impinged at SONGS 1 are anchovy,..."

A review of last three years (1975-1977) of ETS in-plant impingement monitoring reveals that the Queenfish, *Seriphus politus* has been the most predominant species impinged at Unit 1 in terms of both numbers and weight.

Entrainment of anchovy has been sporadic and shows occasional high numbers entrapped probably reflecting the schooling behavior of the species. Early impingement information (pre-ETS-1975) indicating high impingement of anchovy may have been biased by a combination of sampling frequency and these chance occurrences.

It is recommended that the word anchovy be replaced with "Queenfish" to reflect the most recent data available. This change does not effect the overall assessment result indicating no significant effect on recreational or commercial fishing resources.

Offshore Current Induction

Comment 5-32

(page 5-29 paragraph 5)

The applicants agree that there are no detrimental effects of induced circulation on the aquatic environment. However, the discussion of the analysis in the DES concerning the effects of the induced circulation on the aquatic environment should mention that the analysis is based on the diffuser design described in Section 3.4 of the ER-OLS and Section 9.2 of FSAR.

5.5 RADIOLOGICAL IMPACTS

Comment 5-33

(page 5-33, Table 5.4)

Table 5.4 of the DES shows calculated annual doses nearly a factor of 3 greater than the values provided by the applicants in Table 5.2-12 of the Environmental Report - Operating License Stage (ER-OLS). The doses shown in Table 5.2-12 of the ER-OLS were calculated using annual average meteorology.

It appears that the staff has used short term 15th percentile meteorology (valid only for purge releases instead of continuous long-term releases) in calculating the doses shown in Table 5.4 of the DES. The staff is requested to revise the doses consistent with Table 5.2-12 of the ER-OLS.

Comment 5-34

(page 5-34, Table 5.6)

Table 5.6 of the DES shows that the dilution factor used for the dispersion of liquid release is 1. However, Section 5.2.4.3 of the applicants' Environmental Report-Operating License Stage (ER-OLS) shows that the dilution factor is 10 between 0-10 miles and 12.5 between 10-50 miles. The values reported by the applicants were derived consistent with Regulatory Guide 1.112.

The staff is requested to revise the values in Table 5.6 of the DES to be consistent with the dilution factors shown in Section 5.2.4.3 of the ER-OLS.

A-31

5.6 SOCIOECONOMIC IMPACTS

5.6.1 Introduction

Comment 5-35

(page 5-40, paragraph 8)

The second sentence should read:

"The central portion of Orange County ...".

5.6.5 Impact on recreational resources

Comment 5-36

(page 5-44 and 5-45)

The NRC staff concludes in this section and other sections (5.6.5, 9.1, 10.5, and 10.7) of the Draft Environmental Statement (DES), that the applicants' current plan to restrict the public use of the beach in front of the San Onofre Nuclear Generating Station, within the exclusion area, is a significant cost of the project unanticipated at the issuance of the construction permit. Applicants disagree with the conclusion that there will be any significant loss of recreation area.

Subsequent to the issuance of the Final Environmental Statement (FES) required for the construction permits of SONGS 2 and 3, the ASLAB in its initial decision dated December 24, 1974 (ALAB-248) questioned whether recreational activities within portions of the exclusion area should be permitted, and the adequacy of the applicants' authority to control activities in the exclusion area. By Decision dated April 25, 1975 (ALAB-268) the ASLAB ruled that the applicants' authority to control activities within the exclusion area was insufficient and remanded the issue for further hearing.

On October 10, 1975, the applicants submitted Amendment No. 22 to the PSAR consisting of information concerning a proposal for a reduced exclusion area. Amendment No. 22 also provided estimates of the number of persons anticipated within the proposed reduced exclusion area. Applicants' experts estimated the maximum number of persons within the proposed reduced exclusion area would be 31.

The NRC Staff evaluated applicants' assessment of potential beach use as provided in Amendment No. 22 to the PSAR and concluded that applicants' estimates of the maximum number of people on the beach or in the water within the proposed reduced exclusion area were conservative.

The ASLAB Memorandum of Order dated January 22, 1976 (ALAB-308) resolved the issue concerning authority to control activities within portions of the new reduced exclusion area landward of the mean high tide line in the applicants' favor. However, the Board declined to deal with the question concerning the tidal beach and remanded this issue to the ASLB.

The ASLB held hearings on May 19-21, 1976, at which time evidence was heard on several issues concerning the tidal beach, including the anticipated public use of the beach.

Applicants' expert witnesses provided testimony regarding activities within the beach areas in the vicinity of the San Onofre Nuclear Generating Station and the projected number of persons that would be anticipated within the reduced exclusion area. With respect to activities within the beach areas, applicants' expert witness indicated that distances from parking and beach access points to the area in front of the station are such that there will be a low level of activity on beaches within the reduced exclusion area as compared to other beach areas in the San Onofre State Beach because beach users tend to remain relatively close to their point of beach access. With respect to the projected number of persons within the reduced exclusion area, the applicants' expert witness conservatively assumed the total number of persons which could ultimately be accommodated by all park facilities developed to their planned ultimate capacity would occupy the beach at the same time. Based upon a probabilistic distribution of that population , an estimated 35 persons would be located within the reduced exclusion area. Further, based upon actual observations of persons using the San Onofre State Beach, in addition to similar observations on other beaches, it was predicted that the average and maximum number of people using the beach in front of the station, within the exclusion area, would be 7 and 31, respectively.

NRC Staff supported the applicants' contentions and indicated in both written and oral testimony that the area directly in front of the plant was the least desirable both from an aesthetic point of view and for swimming, surfing or sunbathing, and also indicated that when one is laden with beach blankets and other recreational gear, migration up or down the beach would be discouraged, therefore, beach users would congregate relatively close to the paths up the bluffs of the San Onofre State Beach.

ASLB Order dated January 6, 1977, ordered applicant to provide all data collected since March 14, 1976, reflecting the actual daily count of persons using the beach within the applicants' exclusion area, including the tidal beach. Oral Arguments were held on February 1, 1977, during which the applicants' provided an analysis of the daily counts previously submitted to the ASLB. That analysis showed less than 10 persons were observed on the beach in the exclusion area for approximately 57.6 percent of the time, and that, on the average, only 12 to 15 percent of the total number of people observed in the study area (area in front of the station and adjacent areas 1/4 mile north and 1/4 mile south) were in the exclusion area. There was a peak number of 108 persons observed in the exclusion area, however, the 108 persons (40 percent stationary, 19 percent in transit, 20 percent swimming, and 21 percent surfing) represent about 36 percent of the total number observed in the study area. It should be noted that the administrative features proposed in Amendment 22 will only effect stationary persons within the exclusion area. Transit through the exclusion area as well as activities below the mean high tide line such as, swimming, fishing and surfing will remain unrestricted.

The ASLB Initial Decision dated May 20, 1977, ruled in the applicants' favor ordering that the Construction Permits shall be continued in effect.

Given the following facts that:

1. The conclusions drawn by the NRC staff in the DES appear to be based upon the Final Environmental Statement Construction Permit Stage.

2. The ASLAB and ASLB have given detailed consideration, in hearings, regarding usage of the beach in front of the San Onofre Nuclear Generating Station within the exclusion area.

3. The applicants provided expert testimony supporting the fact that the beach in front of the station was the least desirable from the standpoint of aesthetics for swimming, surfing or sunbathing and does not receive significant usage and that people tended to congregate near the paths of the state beach away from the exclusion area.

4. The staff supported the applicants' contention regarding minimal beach usage and undesirability of the beach in front of the station.

In view of the above, the appropriate sections of the DES should be revised to conclude that limiting the use of the beach within the exclusion area boundary and above the mean high tide line to a passage way does not represent a significant loss of recreational space.

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6. ENVIRONMENTAL MONITORING

6.2 PREOPERATIONAL MONTIORING PROGRAM

Comment 6-1

(page 6-2, Fig. 6.1)

The legend is in error, the triangle symbol should represent DO, pH and Heavy Metals. The square symbol should represent continuous temperature.

6.2.1.5 Intertidal Organisms

Comment 6-2

(page 6-3, paragraph 5 and 6)

The monitoring described in the first paragraph was a requirement for Unit 1 which was deleted in September, 1977, because no effects had been detected. Although this study has been deleted as a requirement, SCE has continued an intertidal study program somewhat reduced in scope. The applicants contend continued conduct of this present cobble intertidal sampling program as described below will meet the objectives outlined in the second paragraph of Section 6.2.1.5 of the DES.

The applicants recommend replacing the existing paragrpah with the following paragraph:

"Although not a required component of the monitoring programs, quarterly observations are made along cobble intertidal transects at four monitoring stations and one control station. Predominant macroscopic species and substrate composition are identified and enumerated within three permanent 0.25m^2 (2.69-$ft.^2$) quadrats along a line perpendicular to the beach. Photographs are also taken of each quadrat for a permanent record of ecological changes."

6.2.1.6 Requirements

Comment 6-3

(page 6-3, requirement 2)

The staff requires extensive monitoring of the San Onofre kelp bed based on predictions made in Section 5.4.2.1.

Kelp investigations are currently in progress with the Construction Monitoring Program, which is a special study of the Preoperational Monitoring Program. Detailed methods are outlined in Reference (11). A brief outline of the scope of effort, at all three San Onofre region beds, is as follows:

1. Three benthic stations are located in and about the San Onofre kelp bed and one each at Barn kelp and San Mateo kelp. Stations are quantatively assessed quarterly.

2. Kelp canopies and rock substrate are mapped for areal extent on a quarterly basis.

3. Water nutrient analysis for ammonia, nitrates, nitrites and phosphate taken monthly at all three beds. Water samples are taken for the surface and bottom from within each bed and offshore of each bed. An additional offshore station serves as a monitoring area for upwelling.

4. Kelp tissue analysis for nutrient content is conducted on a monthly basis at all three kelp beds. Each leaf is analyzed for nitrogen content.

5. Assessments of the health of kelp plants in the San Onofre region beds are made on a quarterly basis. Parameters assessed include: success of juvenile recruitment, density of kelp plants, amount of encrusting organisms and grazing by herbivores and abundance of senile and diseased plants.

Based upon the applicants' extensive comments dealing with the predicted impact of the San Onofre thermal plume on the San Onofre kelp bed, the applicants contend that requirement number 2 in Section 6.2.1.6 is unwarranted and should be deleted.

6.3.1 Water quality monitoring program

Comment 6-4

(page 6-6)

The entire section is in error and should be deleted. The program that the staff discusses in the DES is actually a 1976 draft of the applicants' proposed preoperational oceanographic program. An operational program for San Onofre 2 and 3 has not yet been established.

6.3.3 Aquatic biological monitoring

Comment 6-5

(page 6-7, paragraph 2)

This paragraph states, "The applicant intends to forward a description of the study with a schedule for completion to NRC by December, 1978, (see ER, Suppl. 1, p. S1-31)."

In keeping with efforts to avoid duplication and utilize the 316(b) study results, the study plan submittal to the NRC will be made after the completion of the methods development phase of 316(b). We presently anticipate that the 316(b) method development phase will be completed in early 1979, and, therefore, the study plan should be submitted to the NRC by mid-1979.

6.3.3 Aquatic biological monitoring, and

6.3.5 Requirements for Environmental Technical Specifications

Comment 6-6

(page 6-6 and 6-7)

The DES states in Section 6.3.3, paragraph 2 and in requirement number 3, Section 6.3.5, that "...the ichthyoplankton study now being conducted and the required

kelp preoperational program should be continued during operation of the facility until such time as it is possible to state credibly that no significant impacts result from the facility."

The ichthyoplankton study being conducted is a one year program to provide a baseline for comparison with the operational ichthyoplankton study which is also envisioned to be a one year program. Further, as stated in applicants' Comment 6-4, the required kelp preoperational program is considered to be unwarranted and the requirement should be deleted.

A-35

8. NEED FOR THE STATION

8.2 APPLICANT'S SERVICE AREAS AND REGIONAL RELATIONSHIPS

8.2.1 Applicant's service areas

Comment 8-1

(page 8-1, paragraph 2)

The reference number used in the discussion appears to be incorrect.

8.3 BENEFITS OF STATION OPERATION

8.3.1 Minimization of production costs

Comment 8-2

(page 8-3, Table 8.1)

Table 8.1 was derived from the applicants' ER-OLS, Table 1.1-3 and page S.2-188. However, the data found on ER-OLS Table 1.1-3 is not the most current for 1976 and will be updated in a future amendment to the ER-OLS. The applicants have revised Table 8.1 of the DES to reflect changes in data as reported to the Federal Power Commision on Form 1, Annual Operating Report for Southern California Edison Company for the year ending December 31, 1976.
(Revised Table 8.1 (Attachment X))

8.3.2 Energy demand

Comment 8-3

(page 8-4, paragraph 2)

The discussion on the overestimation of peak demands in the 1973 forecast should also mention load management programs. The applicants suggest the last sentence be rewritten as follows:

> "These peak demands were overestimated because the 1973 forecast did not foresee the Arab oil embargo, the following period of economic recession, the nationwide effort to promote energy conservation, and load management."

Comment 8-4

(page 8-4, paragraph 3 and Table 8.3)

The staff's evaluation is based on the 1976 forecast data provided by the applicants in their ER-OLS. The data found on ER-OLS Table 1.4-1[a] is based on an early 1976 forecast and does not reflect the revised forecast (July 23, 1976) data found on ER-OLS Table 1.1-1. SCE has revised Table 8.3[b] of the DES based on ER-OLS Table 1.1-1 and their revised 1976 forecast. The last line in the second paragraph has been changed by the applicants to be consistent with the revised data and reads as follows:

> "SCE's revised 1976 forecast shows a peak demand growth rate of 3.9% from 1976 to 1985, and energy requirements are expected to experience a growth rate of 4.3% in the same period."

a. ER-OLS Table 1.4-1 will be revised in a future amendment to the ER-OLS.

b. Revised Table 8.3 (Attachment Y).

Comment 8-5

(page 8-5)

The discussion of the three forecasts that states, "their projections do not reflect non-price-induced conservation...," this does not consider current SCE forecast methodology. Non-price-induced standards were incorporated into SCE's peak demand forecasts, e.g., the peak demand for 1985 includes a 2.4% reduction due to load management and the "weather sensitive demand" for 1985 was reduced 29% because of building insulation and air conditioning efficiency standards (Reference 19 and 20). Therefore, the discussion on page 8-5, specifically paragraphs 1, 3 and 4 should be modified.
(see Attachments Q and R)

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10. BENEFIT-COST SUMMARY

10.2 BENEFITS

Comment 10-1

(page 10-1, paragraph 2, and page 10-2, Table 10.1)

The net power output for each unit is estimated to be in the range of 1052 to 1106 MWe (see Comment A-1 for discussion). The regional generating capacity will be increased 2104 to 2212 MWe with the addition of Units 2 and 3. The discussion on the primary benefit and Table 10.1 should be revised to reflect the estimated net power output.

10.7 SUMMARY OF BENEFIT-COST

Comment 10-2

(page 10-3, item (2))

The "possible destruction of at least a portion of the San Onofre Kelp Bed during summer months by the heated water discharge" is listed as an additional environmental cost. Because this cost is based on an assessment performed by the staff using disputed data, the applicants request that this cost be deleted if the reassessment of Section 5.4.2.1 Effects of the heat dissipation system warrants such a change.

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UNITED STATES ENVIRONMENTAL PROTECTION AGENCY
REGION IX
215 Fremont Street
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Project # D-NRC-K06002-CA

William H. Regan, Jr., Chief
Environmental Projects, Branch 2
Division of Site Safety & Environmental Analysis
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

FEB 13 1979

Dear Mr. Regan:

The Environmental Protection Agency has received and reviewed the draft environmental statement for the SAN ONOFRE NUCLEAR GENERATING STATION, UNITS 2 and 3, SOUTHERN CALIFORNIA EDISON COMPANY, SAN DIEGO GAS AND ELECTRIC COMPANY.

EPA's comments on the draft environmental statement have been classified as Category ER-2. Definitions of the categories are provided on the enclosure. The classification and the date of EPA's comments will be published in the Federal Register in accordance with our responsibility to inform the public of our views on proposed Federal actions under Section 309 of the Clean Air Act. Our procedure is to categorize our comments on both the environmental consequence of the proposed action and the adequacy of the environmental statement.

EPA appreciates the opportunity to comment on this draft environmental statement and requests three copies of the final environmental statement when available.

If you have any questions regarding our comments, please contact Betty Jankus, EIS Coordinator, at (415)556-6695.

Sincerely,

Sheila M. Prindiville
for Paul De Falco, Jr.
Regional Administrator

Enclosure

Water Quality Comments

1. In Section 5.3.1.1., some assessment is made of the effects of the discharge of heated cooling water on the receiving coastal waters with regards to the California State thermal standards. When evaluating thermal discharge, all effects of Units 2 and 3 should be considered in conjunction with the effects of Unit 1. The natural background is a situation where none of the three units is operating. The natural receiving water temperature as defined by California Thermal Plan (see next paragraph) is "the temperature of the receiving water at locations, depths, and times which represent conditions unaffected by any elevated temperature waste discharge". Unless Units 2 and 3 are not planned to operate concurrently with Unit 1, their effects will occur in concert. All modeling, graphs, and maps produced from models should include Unit 1 effects when evaluating SONGS' effects on the receiving water temperature.

   Under Section 316(a) of the Federal Water Pollution Control Act of 1972 (FWPCA) and under the Water Quality Control Plan for Control of Temperature in the Coastal and Interstate Waters and Enclosed Bays and Estuaries of California (1975 Thermal Plan) (EPA approved State water quality standards), there are several criteria which discharges to coastal waters must fulfill. These should be addressed in any EIS on operating a new coastal discharge of elevated temperature wastes. These are as follows:

   a. In part 3.B.(3.) of the Thermal Plan, it is stated that "the maximum temperature of thermal waste discharges shall not exceed the natural temperature of receiving waters by more than 20°F." Part 3.2.2. of the DEIS states that the cooling water "experiences an 11.1°C (20°F) temperature rise across the condenser." Since the waters in the vicinity of the intakes for Units 2 and 3 are close to the discharge structures for these units, it is possible that these intake waters are already heated beyond their natural temperature. Some evaluation of this effect must be included in the FEIS. The influence of the heated discharge from Unit 1 must also be described. In addition, the intake

and discharge facilities and their depths and how temperature stratification profiles relate to the 20°F requirement should be discussed.

b. In Part 3.B.(4) of the Thermal Plan, it is stated that "the discharge of elevated temperature wastes shall not result in increases in the natural water temperature exceeding 4°F at (a) the shoreline, (b) the surface of any ocean substrate, or (c) the ocean surface beyond 1,000 feet from the discharge system. The surface temperature limitation shall be maintained at least 50 percent of the duration of any complete tidal cycle." Figure 5.3 of the DEIS represents projected incremental increases above natural surface temperatures for the study area. This figure should be changed in the FEIS to include the Unit 1 intake and discharge structures and the increase of surface temperatures already caused by Unit 1 discharges in conjunction with those of Units 2 and 3 so as to compare the increases with the true natural surface water temperature.

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c. In addition, the FEIS should document the estimate (Section 5.3.1.2) of the increase in temperatures at the surface of the ocean substrate around the discharges. This estimate indicates that "violations of the state thermal standards are unlikely." Again, such estimates should compare natural temperatures to the combined effects of Units 1, 2, and 3. These temperatures are of special concern because of the importance of low basal temperatures to maintaining the nearby kelp bed.

d. Finally, the Thermal Plan and Section 316(a) of the FWPCA assert the need to "assure the protection and propagation of a balanced, indigenous population of shellfish, fish, and wildlife in and on the body of water into which the discharge is to be made". In Section 5.4.2.1 of the DEIS, biological/ecological evaluations refer to the effects of the discharges on various types of organisms, indicating the effects to be minimal and acceptable. For plankton, the effects will be "species composition changes" and

"greater respiration rates", also, "significant effects should be localized". For fish, the effects will be mainly "shifts in the types of species (and their numbers) which inhabit the area". For benthic fauna, adverse effects may be expected if "weekly mean temperatures of 22°C prevail for one month or more or where daily temperatures reach or exceed 24°C. It is not, however, anticipated that temperatures averaging 22°C will occur for more than 2 to 3 weeks or that the area experiencing temperatures of 24°C or greater as a result of SONGS operation will be considerably larger than the area experiencing these temperatures under natural conditions". For kelp, the information "suggests that the thermal discharges from SONGS 1, 2 and 3 may result in the destruction of at least a portion of the San Onofre Kelp Bed during the summer months". All of these statements indicate that the indigenous populations will be altered, giving no specific documentation that these effects will be minimal or acceptable. A detailed evaluation of how the aquatic ecosystem will be affected, over what area each species or type of fauna may be influenced, and what constitutes a significant adverse effect should be made and presented clearly in the FEIS.

2. Section 5.4.2.1. <u>Thermal Effects</u>, mentions a final report due on December 29, 1978. This study, provided for under the Thermal Plan and Section 316(a) of the FWPCA, is to be used in evaluating the heat-treatment process which is used to clear the intake facilities of biological growth. EPA considers this study to be an integral part of the assessment of the environmental effects of the thermal discharges from the Units. As such, it must be distributed, along with biological and water quality assessments and conclusions (perhaps in the form of a supplement to the DEIS) to all recipients of this DEIS, with the allowance of a comment period prior to incorporation in the Final EIS.

3. Section 5.4.2.2 includes a discussion of the potential effects of chlorine discharges. The discussion evaluated potential "significant impacts" of the periodic 15-minute chlorine dosing period. The FEIS should include a comparison of effluent concentrations with the State Standards contained in the Water Quality Control Plan for the Ocean Waters of California (1978 Ocean Plan), Table B and Footnote 11, should appear in the EIS. Should the comparison predict that the discharges exceed the requirements, the plans to lower the discharge concentration to agree with the State Standards must be described in the FEIS.

4. No assessment appears in the DEIS of the potential seismic effects of nearby faults on the units, although there is a fault within a mile of the plant (the Christianitos Fault and others in the vicinity). The FEIS should address the potential of seismic events and the resultant damage from fault movement, with particular emphasis on the water quality and off-site radiological contamination.

Radiological Comments

Beach Regulation

This DEIS gives little information on the anticipated beach population. The presence of thousands of daytime beach users and hundreds of overnight campers within 1.5 miles from the reactors has significant security, emergency planning, and radiation dose implications. Consequently, we believe this issue warrants a thorough discussion in the Final EIS so that those reviewers who will not read the Environmental Review and Emergency Plan will be aware of this situation and have an opportunity to evaluate it.

We agree with the decision to restrict usage of the beach in front of the reactors since it will simplify the security and emergency planning problems and will reduce the radiation doses to the population from routine release. However, the practical effectiveness of this restriction should be addressed in the FEIS (e.g., is the prohibition against restricting the area seaward of mean high water, coupled with permitting viewing and pedestrian passage going to make enforcement difficult?).

It would be helpful to briefly mention the Emergency Response Plan that is in effect for the Nuclear Station and relate it to the transient population.

As mentioned under the Dose Commitment section, it is not clear whether beach users and Visitor Center users are included in the individual and population dose calculations.

Environmental Dose Commitments

Page 5-31-34 of the DEIS:

The estimated maximum individual dose and the population dose were independently checked by EPA with results similar to those presented in the DEIS. However, we do have several questions about assumptions used in the DEIS calculations. The FEIS should clarify the following items:

1. The manner in which the individual and population dose to users of the beach is calculated is unclear. For example, what allowance is made for direct radiation doses, especially to those using the walkway between the south and north beaches, and to those at the Visitors Center? Do the individual and population doses include these users of the beach and the Visitors Center and, if so, what assumptions were made on hours of exposure, shielding factors, etc.? Also, it would be helpful if the habits of "a maximum individual" were described so it could be determined to what extent these various pathway dosages are additive.

2. The actual maximum individual dose from present operation of Unit 1 should be described. This dose should be added to those being projected for Units 2 and 3 (from all pathways). This, in turn, should be compared with the 25 millirem per year limit (75 millirem per year to the thyroid) of the Uranium Fuel Cycle Standard (40 CFR 190).

EPA is encouraged that the NRC is now calculating annual population dose commitments to the U.S. population, which is a partial evaluation of the total potential environmental dose commitments (EDC) of H-3, Kr-85, C-14; iodines and "particulates." This is a big step toward evaluating the EDC which EPA has urged for several years. However, it should be recognized that several of these radionuclides (particularly C-14 and Kr-85) will contribute to long-term population dose impacts on a world-wide basis, rather than just in the U.S. To the extent that the draft statement (1) has limited the EDC to the annual discharge of these radionuclides, (2) is based on the assumption of a population of constant size, and (3) assesses the doses during 50 years only following each release, it does not fully provide the total environmental impact. Assessment of the total impact would (1) incorporate the projected releases over the lifetime of the facility (rather than just the annual release), (2) extend to several half-lives or 100 years beyond the period of release, and (3) consider, at least qualitatively or generically, the world-wide influences on the total environmental impact or specify the limitations of the model used.

Environmental Monitoring

The pre-operational and operational radiological environmental monitoring program (as described in Section 6.1.5 of the Environmental Report) appears adequate with the following exceptions which the FEIS should address:

1. A delay of 8 days before analyzing charcoal filter air samples would permit over 99% of the Iodine-133 and 50% of the Iodine-131 to decay before being counted. The decay would be much greater for contamination occurring at the beginning of the 7-day sampling period. The maximum time before analyzing filters should be shortened significantly in order to detect as many incidences of sporadic contamination as possible.

2. It is not clear why a minimum of only ten 7-day air particulate samples are required per quarter. The intent should be to monitor all 13 weeks in a quarter.

3. No TLD stations are indicated for the walkway along the seawall or the mean high water exclusion area in front of the reactors. It would be desirable to include TLD's at these locations to monitor the direct radiation at a site boundary where the public has access.

Reactor Accidents

The EPA has examined the NRC's analyses of accidents and their potential risks. The analyses were developed by NRC in the course of its engineering evaluation of reactor safety in the design of nuclear plants. Since these issues are common to all nuclear plants of a given type, EPA accepts NRC's generic approach to accident evaluation in the DEIS. However, the NRC is expected to continue to ensure safety through plant design and accident analyses during the licensing process on a case-by-case basis.

In 1972, the AEC initiated an effort to examine reactor safety and the resultant environmental consequences and risks on a more quantitative basis. The final report of this effort was issued in October 1975 by the U.S. Nuclear Regulatory Commission as the Reactor Safety Study, WASH-1400 (NUREG-75/014). The EPA's review of this study

included in-house and contractual efforts, and our comments were released in a report in June, 1976. In subsequent discussion with NRC we determined that of the concerns we expressed, those having the most significance with regard to the results of the study were on (1) the latent cancer health effects and (2) the probability of BWR scram failure where we differed by factors of four and a maximum of ten, respectively. We believe that the methodology of the Reactor Safety Study should continue to be used as a tool in the evaluation of nuclear systems that vary from the models chosen for the study, and that a generic analysis should be made of the acceptability of the present risks and the necessity for increased levels of safety.

High-Level Waste Management

The techniques and procedures used to manage high-level radioactive wastes will have an impact on the environment. To a certain extent, these impacts can be directly related to the individual projects because the spent fuel from each new facility will contribute to the total waste. The AEC, on September 10, 1974, issued for comment a draft statement entitled "The Management of Commercial High-Level and Transuranium-Contaminated Radioactive Waste" (WASH-1539). In this regard, EPA provided extensive comments on WASH-1539 on November 21, 1974. Our major criticism was that the draft statement lacked a program for arriving at a satisfactory method of "ultimate" high-level waste disposal. At present, DOE is preparing a new draft statment which will discuss waste management and emphasize ultimate disposal in a more comprehensive manner. EPA concurs with this decision and will review and comment on the new draft statement replacing the September 10, 1974 version when it is available.

EPA is cooperating with both NRC and DOE to develop an environmentally acceptable program for radioactive waste management. In this regard, on November 15, 1978, EPA issued proposed environmental radiation protection criteria (43 FR 53262) for the management of all radioactive waste and will propose environmental radiation protection standards for high-level waste in 1979.

Transportation

In its earlier reviews of the environmental impacts of transportation of radioactive material, EPA agreed with AEC that many aspects of this program could best be treated on a generic basis. The NRC has codified this generic approach (40 FR 1005) by adding a table to its regulations (10 CFR Part 51) which summarizes the environmental impacts resulting from the routine transportation of radioactive materials to and from light-water reactors. These regulations permit the use of the impact values listed in the table in lieu of assessing the transportation impact for individual reactor licensing actions if certain conditions are met. Since San Onofre appears to meet these conditions and since EPA agrees that the routine transportation impact values in the table are reasonable, the generic approach appears adequate for this plant.

The impact value for routine transportation of radioactive materials has been set at a level which covers 90 percent of the reactors currently operating or under construction. However, the basis for the impact, or risk, of transportation accidents is not as clearly defined. At present, EPA, DOE, and NRC are each attempting to more fully assess the radiological impact of transportation risks. The EPA will make known its views on any environmentally unacceptable conditions related to transportation. On the basis of present information, EPA believes there are no unique characteristics of the San Onofre site which would result in greater accident risks than from the "typical" site being studied generically.

Fuel Cycle and Long-Term Dose Assessments

EPA is responsible for establishing generally applicable environmental radiation protection standards to limit unnecessary radiation exposures and radioactive materials in the general environment resulting from normal operations that are part of the total uranium fuel cycle as well as those of the facilities. The EPA has concluded (in 40 CFR 90) that environmental radiation standards for nuclear power industry operations should take into account the total radiation dose to the population, the maximum individual dose, the risk of health effects attributable to these doses (including the future risks arising from the release of long-lived radionuclides to the environment), and the effectiveness and costs of effluent

control technology. EPA's Uranium Fuel Cycle Standards are expressed in terms of dose limits to individual members of the general public and limits on quantities of certain long-lived radioactive materials released to the general environment.

A document entitled "Environmental Survey of the Uranium Fuel Cycle" (WASH-1248) was issued by the AEC in conjunction with a regulation (10 CFR 50, Appendix D) for application in completing the cost-benefit analysis for individual light-water reactor environmental reviews (39 FR 14188). This document is used by NRC in draft environmental statements to assess the incremental environmental impacts that can be attributed to fuel cycle components which support nuclear power plants.

Recently, the NRC decided to update the WASH-1248 survey. We believe this is a prudent step and commend the NRC on initiating this update. In providing comments to the NRC on this subject, dated November 14, 1978, we encouraged NRC to express environmental impacts in terms of potential consequences to human health, since for radioactive materials and ionizing radiation the most important impacts are those ultimately affecting human health. We believe the presentation of environmental impact in terms of human health impact fosters a better understanding of the radiation protection afforded the public.

A second major concern of EPA deals with the discharge and dispersal of long-lived radionuclides into the general environment. In the areas addressed in WASH-1248, there are several cases in which radioactive materials of long persistence are released into the environment. The resulting consequences may extend over many generations and constitute irreversible public health commitments. This long-term potential impact should be considered in any assessment on health impact. EPA has consistently found inadequate the NRC's estimates of population doses for these persistent radioactive materials. In particular, the NRC has generally limited their analysis to the population within 50 miles of a facility or, in rare cases, to the U.S. population, and to doses committed for a 50-year period by an annual release. These limitations produce incomplete estimates of environmental impacts and underestimate the impact in some cases, such as from releases of tritium, Krypton-85, Carbon-14, Technetium-99, and Iodine-129. The total impact of these

persistent radionuclides should be assessed, qualifying such estimates as appropriate to reflect the large uncertainties. In this regard, we note that NEA is addressing this approach in making assessments and that NRC is represented in this effort.

Another major consideration in updating WASH-1248 is the health impact from Radon-222 from the uranium mining and milling industry. Estimates made by EPA, among others, indicate that Radon-222 contributes the greatest fraction of the total health impact from nuclear power generation. In preparing an updated WASH-1248, we believe NRC should:

1. include the Radon-222 contribution from both the uranium mining and milling industries;
2. determine the health impact to larger populations, not only the local populations;
3. recognize the persistent nature of the Radon-222 precursors (Th-230 and Ra-226) by estimating the health impact for a period reflecting multi-generation times.

Decommissioning

The NRC has published a proposed rulemaking on Decommissioning Criteria for Nuclear Facilities in the Federal Register on March 13, 1978. EPA comments were sent to NRC on July 5, 1978, dealing with the decommissioning issue.

In summary, we believe that one of the most important issues in the decommissioning of nuclear facilities is the development of standards for radiation exposure limits for materials, facilities, and sites to be released for unrestricted use. We have included the development of such standards among our planned projects. The work will require a thorough study to provide necessary information, including a cost-effectiveness analysis for various levels of decontamination.

The development of standards for decommissioning must, of course, include consideration of the many concurrent activities in radioactive waste management and radiological protection. EPA has developed proposed Criteria for Radioactive Waste for management of all

radioactive wastes which will provide guidance for decommissioning standards. From the decommissioning view, probably the most important criterion is that limiting reliance on institutional controls (guards and fences) to a finite period. EPA believes that the use of institutional control to protect the public from retired nuclear facilities until they can be decontaminated and decommissioned should be limited at the most to 100 years and preferably less than 50 years. This includes nuclear reactors shut down and mothballed or entombed for a period of time under protective storage. After the allowable institutional care period is over, the site will have to meet radioactive protection levels established for release for unrestricted use. We believe EPA's proposed criteria would be directly applicable, as above, to decommissioning of nuclear facilities and should be given serious consideration by the Nuclear Regulatory Commission (NRC).

The availability of adequate funds when the time to decommission arrives is also most important; it should be the responsibility of the NRC to assure that such provisions are made. We recognize the great complexity of providing funds at construction for decommission in 40 years. However, if it can be determined that the total cost of decommissioning in current dollars is a very small fraction of initial capital costs, provision of escrow funding may not be necessary. Therefore, we urge the NRC to conduct the necessary studies and assessments to determine unequivocally costs of decommissioning and to compare such costs to initial capital costs. It is only through a definitive analysis, and perhaps through realistic demonstrations, that this issue can be resolved.

EIS CATEGORY CODES

Environmental Impact of the Action

LO--Lack of Objections

EPA has no objection to the proposed action as described in the draft impact statement; or suggests only minor changes in the proposed action.

ER--Environmental Reservations

EPA has reservations concerning the environmental effects of certain aspects of the proposed action. EPA believes that further study of suggested alternatives or modifications is required and has asked the originating Federal agency to reassess these aspects.

EU--Environmentally Unsatisfactory

EPA believes that the proposed action is unsatisfactory because of its potentially harmful effect on the environment. Furthermore, the Agency believes that the potential safeguards which might be utilized may not adequately protect the environment from hazards arising from this action. The Agency recommends that alternatives to the action be analyzed further (including the possibility of no action at all).

Adequacy of the Impact Statement

Category 1--Adequate

The draft impact statement adequately sets forth the environmental impact of the proposed project or action as well as alternatives reasonably available to the project or action.

Category 2--Insufficient Information

EPA believes that the draft impact statement does not contain sufficient information to assess fully the environmental impact of the proposed project or action. However, from the information submitted, the Agency is able to make a preliminary determination of the impact on the environment. EPA has requested that the originator provide the information that was not included in the draft statement.

Category 3--Inadequate

EPA believes that the draft impact statement does not adequately assess the environmental impact of the proposed project or action, or that the statement inadequately analyzes reasonably available alternatives. The Agency has requested more information and analysis concerning the potential environmental hazards and has asked that substantial revision be made to the impact statement.

If a draft impact statement is assigned a Category 3, no rating will be made of the project or action, since a basis does not generally exist on which to make such a determination.

Marvin I. Lewis
6504 Bradford Terrace
Phila. PA 19149
3-6-79.

Director, Division of Site Safety Environmental Analysis
Office of nuclear Reactor Regulation
USNRC
Washington. /d.C. 20555

Sir:
NUREG 0490 does a lot of things , but it does not in any way justify the operation of the San Onofre Nuclear Generating Station.

Although the NUREG does provide a lot of good information, this information actually contradicts the usefulness of the SONGS, San Onofre Nuclear Generating Station. For instance, the growth rate in Table 2.2 ,Page 2-2, is 3.5 % or less for the period 1976 to 1990. The growth rate in Table 8.3 and 8.4 on Pages ~~tt~~ 8-4and 8-5 is close to 5% for the same period. Inotherwords , the growth rates in various parts of the report are 'selected ' to provide justification for whatever the writer wishes to fustify in any [illegible] partilular part of the report. This technique is called 'fiction'.

In Appendix D-23 Page 2.5 Seismology is dismissed in a few paragraphs. Considering the recent and continuing ~~xix~~ seismic discoveries at the Hosgri fault at Diablo Canyon (which is in a similar -in fact same- gelological domain), passing off seismology this cavalierly ~~ix~~ is indefensible.

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Page 5-37. First you state in a Table that the Commissioner has directed that Radon 222 will be reconsidered elsewhere; then, the Staff includes Radon 222 in this Nureg in a convoluted and artificial manner which does not in any way investigate or acknowledge Radon 222's full period of toxicity as required by NEPA.

Page 5-39 Tailings are not required to be stabilized forever, and even if it ~~xxxxx~~ were required , forever stabilization is a God like requirement which may be impossible to mortal men.

Chapter 7. This is based entirely on the Rasmussen Wash 1400 . Commissioner Kennedy has already stated on October 18, 1978, "It (Rasmussen Report) found some deficiencies which suggest that the absolute values of the risks presented in the Study should not be used uncritically either in the regulatory process or for public policy purposes."

The DES for operation of SONGS proves unequivocally that this nuclear power plant is unnecessary and dangerous. This is despite the Stall evaluation which ignores all important negative effects.

DO NOT LICENSE THIS NUCLEAR POWER PLANT TO OPERATE AT THE EXPENSE

OF HUMAN LIVES.

Marvin I. Lewis

Southern California Edison Company

SCE

P. O. BOX 800
2244 WALNUT GROVE AVENUE
ROSEMEAD, CALIFORNIA 91770

J. H. DRAKE
VICE PRESIDENT

TELEPHONE
213-572-2288

April 6, 1979

Director, Office of Nuclear Reactor Regulation
Attn: Wm. H. Regan, Jr., Chief
Environmental Projects Branch 2
Division of Site Safety and
Environmental Analysis
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Gentlemen:

Subject: San Onofre Nuclear Generating Station
Units 2 and 3
Docket Nos. 50-361 and 50-362

Mr. Oliver Lynch, Jr., of the NRC staff called on March 27, 1979, to request clarification of Applicants' Comment 6-4 to the Draft Environmental Statement for San Onofre Nuclear Generating Station, Units 2 and 3. Applicants' Comment 6-4 was submitted with other comments by letter to you dated February 2, 1979.

In response to Mr. Lynch's request, a revised Comment 6-4 is enclosed for your information. If you have additional comments regarding this comment, please contact me.

Sincerely,

Enclosure

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6.3.1 Water Quality Monitoring Program
Comment 6-4 (Revised April 6, 1979)
(Page 6-6)

The first five paragraphs of this section of the DES describe a proposed operational monitoring program which was presented in the ER-OLS (Section 6.2) and was based upon the proposed preoperational monitoring program also presented in the ER-OLS. The ER-OLS was developed in 1976 and submitted in 1977 to the NRC.

Since that time, the Preoperational Monitoring Program has been revised to incorporate the latest site specific study results and recent developments in marine ecological study techniques. The revised Preoperational Monitoring Program was approved by the NRC and implemented in 1978. It is the Applicant's intention to develop an operational monitoring program which incorporates results of the Preoperational Monitoring Program and submit it in the near future for approval. It was the intention of Comment 6-4 to indicate that the specific details of the operational monitoring program proposed in the ER-OLS in 1976 (and contained in the DES) should not be considered to represent the program which will actually be implemented. While the program which will ultimately be implemented will be similar to the one included in the ER-OLS, it will not be identical, and the differences between the two cannot be specified at this time because the development process is still underway.

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Attorney for Intervenors

UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of SOUTHERN CALIFORNIA EDISON COMPANY, *et al.*, (San Onofre Nuclear Generating Station, Units 2 and 3)	Docket Nos. 50-361 OL 50-362 OL COMMENTS ON DRAFT ENVIRONMENTAL STATEMENT - SAN ONOFRE NUCLEAR GENERATING STATION, UNITS 2 AND 3

We have carefully reviewed the above draft environmental statement in relation to the requirements imposed by Section 102(2)(c) of the National Environmental Policy Act (NEPA) and 10 CFR Part 51 of the NRC Regulations, and have set forth below intervenors' comments on the proposed action and on this draft statement pursuant to 10 CFR Part 51.25. Intervenors find this draft statement inadequate in a) the discussion and assessment of environmental effects, both beneficial and adverse, associated with the operation of the San Onofre Nuclear Generating Station, Units 2 and 3, and b) the discussion and consideration of available alternatives to the proposed action. Intervenors specifically identify the following deficiencies:

1. The evaluation of cooling water discharge impacts is

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inaccurate and misleading. The heated water will very likely result in the destruction of at least a portion of the San Onofre kelp bed during the summer months, the long-term thermal impacts are likely to be severe, and violations of the state standards will occur. On page 5-7 of the DES it is stated: "The staff concludes that although there exists a remote possibility that state thermal standards could be violated by the operation of Units 2 and 3, violations would, at worst, be infrequent and for short periods. There is no evidence in available drift data to indicate that such an occurence would take place during the summer when thermal impacts would be most severe." This conclusion was apparently based on applicants' "worst case" modeling theory; however, in light of recent findings as a result of studies presently being performed by the Marine Review Committee (MRC) at the request of the California Coastal Commission, it has been determined that the state thermal standards will not be met. The following excerpts from the "Supplemental Staff Report And Recommendations - Review of Thermal Requirements For San Onofre Nuclear Generating Station, Units 2 and 3" prepared by the California State Water Quality Control Board staff are appropriate: "The Report of the MRC confirms the previous prediction that, under normal operating conditions, the proposed discharge will violate the 20 degree F temperature differential in the "receiving waters" i.e., waters at the location and depth of the diffusers of Units 2 and 3. This Report notes: '...if the "receiving" waters are defined as in this paragraph, the standards of the State Thermal Plan will probably be exceeded by the operation of Units 2 and 3.' Although the Report indicates that the discharge will "likely" or "probably"

or "may" violate the temperature differential, there really is no question that such violations will occur." (pp. 4-5)

In a hearing for the purpose of interpreting the term "receiving waters" held on December 21, 1978, the California State Water Quality Control Board held that "...the temperature at the intake point does not represent conditions at the receiving waters," (p. 3 of Opinion of Chairman Bryson and Board Member Mitchell) contrary to applicants' requested interpretation. The net result of this ruling is that the state thermal discharge limitation will be exceeded by operation of SONGS Units 2 and 3.

The DES states at p. 5-27 "The greatest threat of SONGS to the long-term survival of the San Onofre kelp bed is the possibility of injury to the basal tissues from which the canopy is regenerated each year...under extreme worst case conditions (e.g., several days with high ambient temperatures and slack currents, and with all the plants operating continuously), destruction of the basal regenerative tissues might result." The DES further states: "...the community (kelp bed), if destroyed frequently, could never achieve a stable state characteristic of other kelp beds in the area. Furthermore, constant temperature increases coupled with added turbidity would be inimical to interim reestablishment...The perennial occurrence of worst case conditions seems highly unlikely and the staff thus concludes that the long-term thermal impacts from normal station operation are not likely to be severe." (p . 5-27) It is clear that since the state thermal discharge limitation will be exceeded during normal operation of SONGS 2 and 3, the staff's conclusion was based on a faulty premise. Dischargers' normal plant operation will result

in continuous high temperature discharge approximating the worst case conditions and resulting in both short and long-term thermal impacts on the San Onofre kelp beds. The DES states at p. 5-27 "It has been rather well established that temperatures above 18-20 degrees C. (64-68 degrees F) cause deterioration of kelp, and the degree of degradation is directly related to the duration of the exposure to these temperatures."

2. The DES is inadequate in its discussion of the 316(a) exception process as related to thermal pollution caused by the proposed action. Section 6.4.1 of the DES discusses the "thermal exception studies" as related only to periodic "heat treatment" to control fouling organisms. The DES fails to consider the 316(a) exception required for continuous high ambient temperature discharges during the normal operations of Units 2 and 3. It is highly likely that a 316(a) exception request will be forthcoming from applicants in light of the recent denial by the California State Water Quality Control Board of applicants' requested interpretation of the term "receiving waters" as used in the State Thermal Plan. Had applicants' interpretation been approved, it would have obviated applicants' need for a 316(a) exception to the requirements of the FWPCA. Because a 316(a) exception is necessary for the operation of Units 2 and 3 in their present design mode, the DES is inadequate for failure to consider the implications, both short and long-term, on the aquatic environment if such an exception is granted. With respect to the maximum temperature of thermal waste discharges, and contrary to the requirements of 10 CFR Part 51.23(c), due consideration was not given to "...compliance of the facility construction or operation

and alternative construction and operation with environmental quality standards and requirements which have been imposed by Federal, State, regional, and local agencies having responsibility for environmental protection, including applicable zoning and landuse regulations and water pollution limitations or requirements promulgated or imposed pursuant to the Federal Water Pollution Control Act."

3. The DES is inadequate in its evaluation and analysis of the social and economic impact of operating SONGS 2 and 3.

A. With respect to the environmental impact of SONGS on recreational resources, the DES recognizes the failure of applicants to comply with the terms and conditions of the construction permit: "The current plan to restrict the use of approximately 25% of the 3 1/2 mile San Onofre Beach for the 30-year operating life of the plant is a significant loss of valuable recreational and scenic space and represents a substantial change in action between issuance of the FES-CP and application for an operating license." (Section 5.6.5) Staff reiterates previous statements made in the FES-CP that "the beach...is considered to be a unique and scarce recreational resource," (FES-CP, p. 2-11) and "that closure even for a brief period is objectionable" (FES-CP, p. 8-11). Despite the re-affirmation of these judgments, staff concludes that the social and economic impact of operating SONGS 2 and 3 - with the significant exception of restricting public use of the beach - will be only "moderate". The overall impact will be more severe than "moderate" if the beach access restriction is factored into the balancing process. Staff's treatment of this issue is misleading and inconsistent

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with the purpose and intent of NEPA, section 102(2)(c), which calls for preparation of a detailed statement on, among other things, any *irreversible and irretrievable commitments of* resources which would be involved in the proposed action should it be implemented. Restriction of the public's use of this beach *is* such an irreversible and irretrievable commitment of resources.

B. With respect to the economic impact of SONGS 2 and 3, the DES provides no analysis of the effects of the Jarvis-Gann Amendment (Proposition 13). The DES states that "The applicant should reassess the potential tax benefits accruing to these jurisdictions and districts in light of Proposition 13." (p . 5-44) This is a wholly inadequate treatment of the economic impact of SONGS 2 and 3, inasmuch as the revenue from the plant and its allocation within communities will be "significantly different from what was assumed" - to use the staff's own words - in this economic impact analysis. (p . 5-44, section 5.6.4)

4. The DES inadequately evaluates the environmental impact of postulated accidents in that Class 9 occurrences were omitted from consideration. (Section 7-1) The DES states on p. 7-2 with respect to Class 9 occurrences that "Their consequences could be severe." The DES fails to discuss the probability of Class 9 occurrences in a complete and comprehensive manner. In view of the recent earthquake fault discoveries near the San Onofre site and the existence of the dewatering-well cavities found beneath the site, a full discussion of failures more severe than *those* required for consideration in the design bases of protective systems and engineered safety features (Class 9) is warranted. Further, the *estimated dose of* 1400.00 *man-rems to* population in

the 50-mile radius for a large-break loss of coolant accident (Table 7.2, p. 7-3, Class 8.1) is substantial and inadequately discussed, if at all, in the text.

5. The DES is inadequate in that it fails to discuss the environmental impacts to the region in the event of an accidental release of radiation requiring evacuation. No discussion is contained in the DES as to the adaptability of the San Onofre site to adequate evacuation processes including evacuation of the nearby beach areas during times of peak use; no discussion is contained in the DES as to the suitability of existing evacuation plans; no discussion is contained in the DES as to the effects which adoption of the NRC/EPA Task Force Report on Emergency Planning (NUREG-0396) will have on evacuation within the new and expanded Emergency Planning Zone as distinct from the presently designated Low Population Zone (NRC Regulations 10 CFR Part 100).

6. The DES is inadequate in that it fails to reassess the seismic design basis for SONGS 2 and 3 in light of a) the dewatering-well cavities and b) the recent earthquakes and faults discovered since the current design basis was established.

7. The DES is inadequate in that the cost/benefit analysis fails to provide consideration for the greatest possible *escalation of uranium prices, based on recent occurrences, for* SONGS 2 and 3 over the operating life of the plant. The projected fuel costs identified as $87,900,000/yr for 1981 (Table 10.1, p. 10-2), will possibly escalate to a prohibitively high level since long-term uranium contracts are generally tied to market price at delivery or 7$ per year escalation, whichever is greater Staff admits (section 10.3) that since the issuance of the FES-CP

the fuel, operating, and maintenance costs of nuclear plant operation have escalated more rapidly than anticipated. The DES does not discuss adequately the possibility of additional future escalation of costs with respect to the fuel requirements of San Onofre, and does not utilize a "worst possible case" approach to determine total fuel costs over the operating life of the plant. The cost/benefit analysis contained in the DES is therefore invalid.

8. The DES is inadequate in that it fails to discuss the possibility that decommissioning costs may escalate to prohibitively high levels by the end of the operating life of the plant, at which time the applicant is required to prepare a proposed decommissioning plan for review by the NRC. (Section 9.4) Although NRC regulations do not require the applicant to have developed a decommissioning plan at the time an operating license is obtained, the discussion of alternative decommissioning methods and their associated costs found in the DES is misleading and does not present an accurate projection of what the actual decommissioning costs for SONGS will be. Staff calculations for determining decommissioning costs per unit of electricity generated do not utilize a start-up date of 1981 or an escalation rate based on the current rate of inflation. Staff's projection that "For the SONGS Units 2 and 3 the decommissioning costs would be about double that indicated for all of the decommissioning one-unit alternatives" (p. 9-17) is wholly inadequate for purposes of making an informed cost/benefit judgment. As a consequence, the cost/benefit analysis for SONGS 2 and 3 is invalid.

9. The DES is inadequate in that it fails to comprehensively discuss the temporary storage of nuclear waste materials, including the interim storage of spent fuel, on site.

10. The DES is inadequate in that it fails to discuss the issue of plant security and provide assurances that all nuclear materials will remain accounted for and protected from the risk of terrorist or criminal activity or sabotage.

Because due consideration was not given to compliance with the requirements of 10 CFR Part 51.23(c), and because this DES fails to consider all environmental impacts of the proposed action and alternatives to the proposed action as required by Section 102(2)(c) of NEPA, staff's conclusion that the action called for is the issuance of operating licenses for Units 2 and 3 of SONGS is premature and founded on insufficient and inaccurate data.

For the foregoing reasons, intervenors request that the NRC either a) adequately address the issues raised above in the final environmental statement for SONGS 2 and 3, or b) deny applicants' request for licenses to operate SONGS 2 and 3.

Dated: Jan 30, 1979 Respectfully submitted,

Richard J. Wharton
RICHARD J. WHARTON
Attorney for Intervenors

COMMENTS ON SUPPLEMENT TO DRAFT ENVIRONMENTAL STATEMENT

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FEDERAL ENERGY REGULATORY COMMISSION
WASHINGTON 20426

IN REPLY REFER TO:

January 23, 1981

Mr. Frank J. Miraglia
Acting Chief, Licesning Branch
No. 3
Division of Licensing
U.S. Nuclear Regulatory Commission
Washington, D. C. 20555

Dear Mr. Miraglia:

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I am replying to your request of January 16, 1981 to the Federal Energy Regulatory Commission for comments on the Supplement to the Draft Environmental Impact Statement related to the operation of the San Onofre Nuclear Generating Station, Units 2 and 3. This Draft EIS has been reviewed by appropriate FERC staff components upon whose evaluation this response is based.

This staff concentrates its review of other agencies' environmental impact statements basically on those areas of the electric power, natural gas, and oil pipeline industries for which the Commission has jurisdiction by law, or where staff has special expertise in evaluating environmental impacts involved with the proposed action. It does not appear that there would be any significant impacts in these areas of concern nor serious conflicts with this agency's responsibilities should this action be undertaken.

Thank you for the opportunity to review this statement.

Sincerely,

Jack M. Heinemann
Advisor on Environmental Quality

810 408

United States Department of Agriculture — Economics and Statistics Service — Washington, D.C. 20250

January 26, 1981

Mr. Frank J. Miraglia
Acting Chief, Licensing Branch No. 3
Division of Licensing
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Dear Mr. Miraglia:

Thank you for forwarding the Supplement to the Draft Environmental Statement for the San Onofre Nuclear Generating Station, Units 2 and 3.

We have reviewed the material, Docket Numbers 50-361 and 50-362, and have no comments at this time.

Sincerely,

MELVIN L. COTNER
Director, Natural Resource
Economics Division

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8102020412

Oceanside, California
February 6, 1981

Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Gentlemen:

I wish to comment on the environmental impact report your organization supplied on the installation and licensing of the nuclear installations at San Onofre which are practically in our back yards:

1.- From what I read in the Oceanside Blade-Tribune no mention was made re. the extreme hazards and the potential of submarines from an enemy action shelling these huge nuclear concrete domes, shattering them thereby releasing that deadly nuclear radiation which could wipe out this entire area. We think this is a tremendous oversight and needs to be studied along with earthquake consideration;

1a.- A remedy in the area of earthquakes and war-time action would be to pipe some of that 100 trillion cubic feet of natural gas that CBS's "60 Minutes" said exists under the U.S. continent. (President Reagan's own people admit to this). And fire the San Onofre steam boilers with the gas and eliminate all of the dangers on which you debate. I pray to God that you consider this alternative and act upon it. People's lives and health are more important than corporate profit.

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2.- We believe that due to the fact that SDG&E will control only 20% of the nuclear output and Southern California Edison will control 80% and by grid move all of this 80% of the generated power to areas that are not threatened by radiation that there should be (if these plants are activated) from 60% to 30% rate discounts for people who live close to this operation. This proposal is now before the Calif. Public Utilities Commission. If you license these plants we would appreciate your recommendation to the PUC. France and Japan presently are allowing this type of discount due to the hazards of nuclear energy. So the proposal does have precedent. 8102110284

Office of Nuclear Reactor Regulation (2) February 6, 1981

3.- To be complete and legally valid we believe that all elements relating to the subject need to be included in the EIR. Apparently the subject of potential enemy action on these nuclear plants was not included and it needs to be discussed.

In closing may I request an answer to the positions expressed in this letter. I will be extremely grateful. I am.

Very sincerely

Frank H. Grundel
1888 Blackhawk St
Oceanside, Calif 92054

Perspective
TIMES-ADVOCATE
SUNDAY, JANUARY 25, 1981 C-1

C-4 TIMES-ADVOCATE, ESCONDIDO, CA., SUNDAY, JAN. 25, 1981

Nuclear neighbor asks for discount

By DICK PHILLIPS
T-A Staff Writer

OCEANSIDE — An Oceanside man is working to achieve a considerable reduction in utility rates for those living near the San Onofre nuclear power plant.

Frank Arundel, 1888 Blackhawk St., proposed the compensation for those residents he thinks live in a danger zone — near San Onofre. He thinks they should get a 60 percent discount in electrical rates.

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Residents within 30 miles should receive 50 percent rate reduction, he says, and those within a 30- to 40-mile radius a 40 percent rate reduction; people living in a 40- to 50-mile radius should have their rates cut by 30 percent.'

"People here are being gougëd to death by utility rates," said Arundel, 73. "With this plan, the next time they build a nuclear plant, they'll put it out of the umbrella of people where it wouldn't be troublesome in case of earthquake or war. They wouldn't put these plants at our backdoor.

"If we have to live here and bear the brunt of nuclear power, we should be beneficiaries of cheap electricity, particularly if 80 percent of the energy produced there will be transmitted outside this area anyway."

Arundel's plan did not impress the Public Utilities Commission, which has indicated there is little chance of seeing the policy implemented statewide.

The "chances of this plan flying are slim" because it would be discriminatory ratemaking, one PUC spokesman said.

San Onofre, 18 miles north of Oceanside, has one operating nuclear plant, which was shut down for repairs through most of 1980. Units 2 and 3 are nearing completion, at a cost of $2.3 to $5 billion. Both are designed to produce 1,100 megawatts of electricity.

Southern California Edison, based in Los Angeles, holds 80 percent interest in the nuclear plant and SDG&E has 20 percent. The SDG&E service area consumes about 2,300 megawatts of electricity.

Martin Mattes, legal adviser to John Bryson, PUC chairman, said copies of Arundel's proposal have been given to the commissioners for study. "But, I don't know of any action planned on the subject," Mattes said.

He said the commission reaches decisions in three ways. Under one, a consumer may apply for a rate change. "This is one way Arundel could intervene and advocate his proposal," Mattes said.

"Or, he could file a complaint against a utility for discriminatory rates, for example. But, the burden of proof is upon the complainant and it's difficult to win a case this way," Mattes said.

The commission can also initiate an investigation into an area of interest. "It's possible the PUC may decide to pursue this and investigate," Mattes said.

In his reply to Arundel, Mattes said he discussed several problems with the discounted rate plan. "If the PUC adopts rate discounts based on unfavorable aspects of having a utility company in the neighborhood, people will make other demands based on similar situations," the adviser said.

For example, those living near an operating fossil fuel plant suffer because of pollution, he said. Transmission lines may be another unfavorable aspect. "The commission is already faced with substantial complications in ratemaking procedures," Mattes said.

Arundel disagrees: "If we're going to put up billions of dollars for these plants and they're going ahead and build them anyway, we should be the beneficiaries." He said 220,000 area residents would fall under the discount plan

Frank Arundel: 'We bear the brunt'

United States Department of Agriculture
Soil Conservation Service
2828 Chiles Road
Davis, CA 95616
(916) 758-2200

February 11, 1981

Mr. Frank J. Miraglia
Acting Chief, Licensing Branch No. 3
Division of Licensing
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Mr. Miraglia:

The Soil Conservation Service has reviewed the Supplement to the Draft Environmental Statement for San Onofre Nuclear Generating Station, Units 2 and 3. We find no controversial items within the realm of SCS responsibilities.

This Environmental Statement Supplement reveals no conflicts with any of the ongoing projects within our jurisdiction. No prime land will be lost to the proposed project.

We appreciate the opportunity to review and comment on this report.

Sincerely,

Francis C. H. Lum
FRANCIS C. H. LUM
State Conservationist

cc: Norman A. Berg, Chief, SCS, Washington, D.C.
Jack Smith, Area Conservationist, Escondido, CA

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United States Department of the Interior

OFFICE OF THE SECRETARY
WASHINGTON, D.C. 20240

ER 81/80

MAR 2 1981

Mr. Frank J. Miraglia
Acting Chief
Licensing Branch No. 3
Division of Licensing
Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Mr. Miraglia:

We have reviewed the supplement to the draft environmental statement for San Onofre Nuclear Generating Station, Units 2 and 3, San Diego, California, and find we have no comments. The opportunity to review this document is appreciated.

Sincerely,

CECIL S. HOFFMANN
Special Assistant to
Assistant SECRETARY

A-55

RICHARD J. WHARTON
Attorney at Law
University of San Diego
Alcala Park, California 92110

(714) 291-6480 Ext. 4376

Attorney for Intervenors

RECEIVED MAR 12 1981 U.S. NUCLEAR REGULATORY COMMISSION

UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of SOUTHERN CALIFORNIA EDISON COMPANY, et al. (San Onofre Nuclear Generating Station, Units 2 and 3)	DOCKET Nos. 50-361 OL 50-362 OL JOINT INTERVENORS COMMENTS ON SUPPLEMEN TO DRAFT ENVIRONMENTAL STATEMENT RELATE TO OPERATION OF SAN ONOFRE NUCLEAR GENERATING STATIONS, UNITS 2 and 3 (NUREG-0490)

The Supplement to Draft Environmental Statement (NUREG-0490, December, 1980), hereinafter referred to as NUREG-0490, prepared by the Office of Reactor Regulation (Staff) of the United States Nuclear Regulatory Commission (NRC) related to the operation of San Onofre Nuclear Generating Station, Units 2 and 3 (SONGS 2 and 3) has been reviewed by Intervenors in relation to the requirements imposed by the National Environmental Policy Act (NEPA) (42 U.S.C. § 4321, et seq.), 10 C.F.R. Part 51, and 40 C.F.R. Part 1502. Intervenors comments on the proposed action and on NUREG-0490 are made pursuant to 10 C.F.R. Part 51.25 and 40 C.F.R. Part 1503.

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The purpose of NUREG-0490 was "to identify and evaluate the site-specific environmental impacts attributable to accident sequences that lead to releases of radiation and/or radioactive materials including sequences that can result in inadequate cooling of reactor fuel and to melting of the reactor core." NUREG-0490, p. vi. These accident sequences are commonly referred to as meltdowns or Class 9 accidents.

The NRC's historic first site-specific impact study of a meltdown accident at a California nuclear reaction is inadequate, incomplete and misleading. NUREG-0490 is misleading because it does not provide decision-makers with sufficiently detailed information regarding the potential environmental impacts of a meltdown at SONGS 2 and 3 to aid them in a substantive decision whether or not to proceed with granting an operating license to this federal nuclear project in light of the economic and other consequences of an accident at SONGS 2 and 3. NUREG-0490 does not encourage public participation because it does not make adequate information available to the public in non-technical language about the potential economic and environmental impacts that could affect the lives of twelve million people. NUREG-0490 appears inadequate and incomplete when compared with other independent meltdown impact analyses.

After the Three Mile Island accident, which resulted in mass evacuations and temporary relocation of many people, the California State Legislature passed a law (Senate Bill 1183, now Section 8610.5 of the Government Code), which required the State

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Office of Emergency Services (OES) to prepare Emergency Response Plans for potentially severe nuclear accidents involving the release of large amounts of radiation. In order to plan for such accidents, the State required information of the potential scenarios and consequences that could result from meltdowns in California reactors. The State lead agency, OES, contracted with a conservative consulting group, Science Applications, Inc. (SAI), to study the consequences and potential scenarios of meltdowns at California reactors. SAI has conducted research for the NRC, the Department of Energy, nuclear military projects, nuclear utilities, and the nuclear industry. The SAI-OES study was released to the public in Sacramento, California on July 15, 1980. The portion of the SAI-OES study which relates to SONGS 2 and 3 was based on extensive site-specific data whereas NUREG-0490, while it purports to be based on site-specific data, considers mainly excerpted "data, methodology and assumptions" from the WASH-1400 study. The inadequacies of this approach are demonstrated by the following comparison between the SAI-OES study and NUREG-0490 consequence analyses:

The SAI-OES study indicates that the maximum consequences for a nuclear meltdown at SONGS 2 and 3 would be $180 billion in economic cost consequences, NUREG-0490 estimates $35 billion; SAI-OES estimates 16,000 square miles of land contaminated with radiation, NUREG-0490 estimates 3,000 square miles; SAI-OES estimates eight to ten million Southern Californians would be required to relocate and leave their homes and property for up to ten years. Four to five million of them would have to be relocated longer than ten

years, NUREG-0490 gives no estimates for the magnitude of the population affected by relocation. SAI-OES estimates that in 1975 there were 7.7 million people living within 60 miles of the San Onofre site. Within 100 miles there are approximately 12 million people. The SAI-OES study acknowledges that "Latent deaths from San Onofre can occur within 100 miles, which includes half of the population of California." Another report done for the California State Legislature, discussed below, warns that children within 100 miles downwind from the reactor would receive damage to their thyroid glands and would require surgery due to exposure to radioactive iodine gases. The SAI-OES study also estimates that $6.6 billion in cost consequences could occur within 500 miles of San Onofre following a meltdown. Reports to the President's Council on Environmental Quality warn that areas as far away as 1,000 miles or more could be affected, and that up to 125,000 square miles of land could suffer some contamination or crop or milk interdiction. The possibility exists that Southern California could be permanently contaminated after a meltdown at SONGS 2 and 3. This is not surprising when we look at other accident scenarios and compare their estimates.

One NRC analysis of reactor accidents, WASH-740, estimated that an area the size of Pennsylvania could be permanently contaminated by a meltdown at a reactor significantly smaller than either Unit 2 or 3 at San Onofre. Another report, the Rasmussen report, WASH-1400, estimated that 3,000 square miles of land would be contaminated, but assumed that effective

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evacuations would take place out to 30 miles downwind from the reactor accident. NUREG-0490, estimates the maximum consequences of a San Onofre meltdown to be $35 billion in costs for mitigating actions (evacuations, relocations, land interdiction, emergency response by local, county, state and federal teams), 1 million people would receive more than 25 rems, there would be 130,000 acute fatalities, and 300,000 latent cancers in the population within 50 miles who would be exposed to 30 to 40 billion person rems released during the accident.

The consequences of nuclear power plant core melt accidents have also been estimated at the request of the California State Legislature and the President's Council on Environmental Quality by Dr. Jan Beyea and Dr. Frank von Hippel, nuclear physicists with the Princeton University's Program on Nuclear Policy Alternatives of the Center for Energy and Environmental Studies. Dr. Beyea noted in his analysis that a meltdown with a release of radioactive gases from a large reactor could involve "health effects and possible land use restrictions have been considered out to distances of 1,000 miles and for periods of decades after the release." He estimates that up to 175,000 square miles of land could be under some form of interdiction or restricted use following the meltdown. He explains this by saying "The number of health effects and the . . . land contamination can range so high because a substantial fraction of the released radioactivity can be carried for hundreds of miles downwind

before being removed from the atmosphere by deposition on the ground. Dr. Beyea told the President's Council on Environmental Quality (CEQ) that "early fatalities could occur up to 30 miles downwind" of a reactor meltdown. Dr. Frank von Hippel testified before the California State Legislature after Three Mile Island that "the thyroid could receive a radiation dose tens to hundreds of times higher than the rest of the body. Exposed children more than a hundred miles downwind would suffer thyroid damage which would require surgery years later." (emphasis added) NUREG-0490 did not reference the SAI-OES study, in spite of the fact that the Atomic Safety and Licensing Board (ASLB) and the NRC Staff were made aware of the report by Intervenors during July and August of 1980, six months before NUREG-0490 was issued.

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The SAI-OES study is a conservative report in that it calculates its predictions and models based on site-specific data. NUREG-0490 is not conservative and is inadequate because it is not sufficiently based on site-specific data. The SAI-OES report used extensive site-specific data regarding the nearby population centers and the various weather conditions in Southern California. That report identified several site-specific unique features which should have warranted a different conclusion from the NRC Staff than "there are no special or unique features about the San Onofre site and environs that would warrant special or additional engineered safety features for the San Onofre plants." Joint Intervenors conclude there are special and unique features that exist at the San Onofre site which are listed as follows:

(1) The three reactors at San Onofre are uniquely located near the intersection of two major Fault Zones, the Cristianitos and the Newport-Inglewood. Prior to 1980, the NRC believed there was no structural relationship between the two Fault Zones. However, in 1980, federal and state marine geologists discovered a new zone of faults which they named "Cristianitos Zone of Deformation" which project directly beneath the three reactors. Thus, the possibility of damage to the reactors during earthquakes is higher now because of the possibility of surface rupture directly under the reactors. This was not factored into the Rasmussen Report, WASH-1400, the Lewis Report, SAI-OES or NUREG-0490. NUREG-0490 does not even mention geologic-seismic site-specific events as a significantly possible factor in the probabilistic risk assessment.

(2) The San Onofre site is uniquely located on the Pacific plate, near the Plate Tectonic Boundary Fault, the San Andreas. San Onofre is moving north in relation to the North American Plate. These reactors are uniquely migrating north on a geologic time scale. Plate Tectonics were not understood when the San Onofre site was originally chosen in 1962. It was not until 1969 that the plate tectonics theories were accepted.

(3) The San Onofre site has the unique feature of being sited close to San Onofre Unit 1. If Unit 1 had a meltdown, it would severely affect operations of Units 2 and 3, resulting in various consequences, none of which were considered in NUREG-0490. The older reactor at the site, San Onofre Unit 1,

was identified by the SAI-OES analysis as having the highest probability of a meltdown of any reactor in California for two primary reasons. "The first reason is that the Unit One auxiliary feedwater system depends on operators to align and initiate the system. Potential failures due to human factors make the system less reliable than automated systems. The second reason relates the long term recirculation mode of emergency core coolant, which requires at least one of two pumps located in the containment. In the event of a pump failure, repairs cannot be made because the pump is inside the containment and would be isolated during an accident." NUREG-0490 does not consider the proximity of SONGS 2 and 3 to Unit 1 to be a unique or special feature.

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(4) San Onofre Unit 1 has been shutdown for approximately one year due to leaky corroded steam generator tubes. The NRC issued a report in 1976 (NUREG-0900-5, Report to Congress on Abnormal Occurrences) which explained that "The failure of a number of steam generator tubes as a result of the pressure transients during a loss of coolant accident could render the emergency core cooling system ineffective." The Unit 1 was not designed for the magnitude of ground motions that Units 2 and 3 were. An earthquake could conceivably only damage Unit 1, because of its structurally weak steam generator tubes, but that could result in a LOCA (loss of coolant accident) and a meltdown, which would affect the two other reactors and the environment.

(5) The San Onofre reactors are special and unique in that the reactor core of Unit 2 was installed backwards, necessitating total rewiring of the control room and other systems.

(6) The San Onofre site is unique also in that San Onofre Unit 2 was constructed above earthquake faults that were not discovered until 1974 during construction excavations.

(7) SONGS 2 and 3 are underlain by dewatering cavities that developed during construction. Intervenors believe this also is a special of unique feature at SONGS 2 and 3 which must be considered.

(8) The Southern California region, including San Onofre, frequently has weather inversions. During these inversions, air pollutants, including accidentally leaked radioactive gases, can be trapped beneath the inversion layer, where they can only mix and travel horizontally. Thus, a meltdown at SONGS 2 and 3 could affect the nine to ten million people who live in the air basins that share the same East Pacific high pressure zone inversion layers. Although NUREG-0490 admits that "accident consequences are very much dependent on the weather conditions existing at the time . . ." they do not specifically consider the unique Southern California high pressure inversion layers which are a predominant characteristic of the San Onofre site.

(9) The San Onofre reactors are uniquely located on a Southern California beach state park that stretches for many miles, but which is inaccessible and inescapable except by driving past the reactors on the old-highway, now running parallel to Interstate-5. On a typical summer day, 25,000 persons drive close to the reactors on a narrow and curving road. These beach-goers could be trapped during a meltdown, especially if

an earthquake occurred at the same time or caused it.

(10) Another unique or special feature of San Onofre is its proximity to roads used by thousands of uncontrolled travelers per day which presents a unique possibility for sabotage accidents that could lead to releases of radioactivity.

(11) The San Onofre site is special and unique in that one-half of the population of the State of California lives within 100 miles of the site.

(12) It is a unique feature of SONGS 2 and 3 to be the largest reactors ever considered for operating licenses.

(13) The San Onofre site is unique in that it is sited within contamination distance of a major portion of the nation's fresh produce farms, especially in the winter months.

(14) The San Onofre site is also unique in that it could cause international economic and environmental impacts by contamination of a significant part of Baja California's agricultural resources.

After the Kemeney Commission and the Rogovin Report were issued on Three Mile Island, the Council on Environmental Quality wrote a letter to the Nuclear Regulatory Commissioners on March 20, 1980. The letter released the results of the CEQ review and critized the NRC's lack of compliance with NEPA laws in the EIS analyses of potential accidents at reactors. The CEQ stated that the NRC's EIS discussions of "potential accidents and their environmental impacts was found to be largely perfunctory, remarkably standardized, and uninformative to the public." The CEQ also advised the NRC that "site specific treatment of data

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should be substituted for "'boilerplate' assessment of accident initiating events and potential impacts, and EIS's should be comprehensible to non-technical members of the public..." Intervenors comment upon the fact that NUREG-0490 contains 29 pages of text with about 8 pages of site-specific information which is selective and slanted. NEPA requires detailed statements of aspects of proposed action significantly affecting the quality of the human environment and Intervenors feel NUREG-0490 is inadequate in that it is"largely perfunctory, remarkably standardized and uninformative to the public."

NUREG-0490 is also inadequate in that it failed to consider earthquake induced core melt accidents. While the Reactor Safety Study(RSS), WASH-1400, concluded that the probability of core melt accidents in nuclear power plants from seismic events was insignificant compared to core melt probabilities from other accidents, recent assessment of the potential for earthquake induced core melt accidents suggests that the probability of such events may be significant when compared to core melt accidents from other causes considered by RSS. Intervenors contend that the seismic design basis for SONGS 2 and 3 is inadequate and, therefore, consider it prudent to evaulate the potential for seismic-induced core melt accidents at SONGS 2 and 3 to establish if they may be significant factors. The purpose of NUREG-0490 was to identify and evaluate site-specific environmental impacts. It does not evaulate the potential for seismic-induced core melt accidents and, therefore its probabilistic assessment of risk at SONGS 2 and 3 is inadequate.

NUREG-0490 is further inadequate and particularly misleading in its assessment of health affects avoidance (Section 7.1.1.4). NUREG-0490 did not mention thyroid blocking in its assessment of health affects avoidance, relying only on restriction of contaminated property and foodstuffs. Dr. Frank von Hipple in his testimony before the California State Legislature states:

> The thyroid can be protected against absorbing radioiodine, however, if before the cloud arrives you take about one thousand times your ordinary daily iodine intake in the form of potassium iodide (the form of iodine present in iodized salt). This will saturate the thyroid with ordinary iodide and reduce its ability to absord the radioactive iodide when it arrives. This strategy was recommended in the American Physical Society's reactor safety study four years ago. The Food and Drug Administration approved potassium iodide for emergency thyroid 'blocking'. . . I would recommend that California do two things with regard to this thyroid protection strategy:
>
> 1) Develop a stockpile of potassium iodide in the appropriate dosage in either sealed foil wrapped pills or liquid solution. This would not be costly. Based on a 1972 study for the Defense Civil Preparedness Study, it appears that enough pills for the entire nation could be produced for a few million dollars.
>
> 2) The more difficult part of the job would be to develop an effective distribution system. If one waited until a cloud of radioiodine had been released before distributing the blocking chemical and informing the public of its use, one might well be too late. (A week after the beginning of the crisis at Three Mile Island, the Pennsylvania state government refused to distribute the chemical to the population within 10 miles of the site - despite the joint recommendation to do so from the Surgeon General, the Food and Drug Commissioner, and the Director of the National Institutes of Health who thought that sufficient warning time might not be available to protect this population in case a release occurred. On the other hand, if people were given potassium iodide to keep in their medicine cabinets along with asprin, it is likely that many would lose track of it pretty quickly. Perhaps it should be attached by the local utility to household electricity meters and its presence announced in case of need. The best strategy is obviously a problem well worth a study. California could break some important ground here."

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Section 7.1.1.4. is particularly misleading in its statement that "radiation hazards in the environment tend to disappear by the natural process of radioactive decay (but) can continue for a relatively long period of time -- months, years or <u>even decades</u>." (emphasis added) This misleading statement fails to note that some ratioactive wastes from nuclear accidents such as radioactive Strontium and Cesium can enter the food chain and remain a hazard for 1,000 years or more. Other isotopes remain a hazard for 1 million years or more.

NUREG-0490, Section 7.1.3. entitled Mitigation of Accident Consequences is inadequate in that it fails to note that consequences could be reduced by retrofitting SONGS 2 and 3 with filtered venting systems to prevent accidental releases of radioactive gases.

NUREG-0490, Section 10 is misleading, inadequate and incomplete. The Section contains three sentences with regard to its conclusions and Re-Evaluated Benefit-Cost Balance. This section should be expanded because the environmental risks of a Class 9 accident involve the entire region of Southern California, Norther Baja California, Mexico, and parts of Arizona. These regions could be permanently contaminated with radiation following a coremelt at SONGS 2 and 3. The risks involve the

value of all real and personal property, both public and private in those regions. The risks involve fatalities, latent cancer deaths and genetic damage. The risks involve compensation to victims in the event of such accidents. Section 10 of NUREG-0490 concludes that the environmental risks of Class 9 - coremelt accidents - "does not change the results of the cost-benefit balance contained in the Draft Environmental Statement (Section 10)."

CONCLUSION

NUREG-0490 concludes "that there are no special or unique features about the San Onofre site and environs that would warrant special or additional engineered safety features for the San Onofre plants." Intervenors conclude there are unique characteristics at SONGS 2 and 3 that warrant additional engineered safety features especially in light of the unique earthquake hazard which could cause a coremelt accident and common-cause failure of essential safety systems at SONGS 2 and 3. A future earthquake near the San Onofre site could be the common cause for failure of the cooling systems of all three reactors on the San Onofre site and all three of the spent fuel pools simultaneously. This would be the worst case accident that should be analyzed by the NRC and this analysis should be a part of a revised NUREG-0490.

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CERTIFICATE OF SERVICE

I hereby certify that the JOINT INTERVENORS COMMENTS ON SUPPLEMENT TO DRAFT ENVIRONMENTAL STATEMENT RELATED TO OPERATION OF SAN ONOFRE NUCLEAR GENERATING STATIONS, UNITS 2 AND 3 (NUREG-0490) have been served on the following by deposit in the United States mail, first class, postage prepaid, this 9th day of March, 1981:

Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D. C. 20555

Attention: Director, Division of Licensing

Executed on March 9, 1981 at San Diego, California.

Darity Wesley
DARITY WESLEY

Union of
CONCERNED
SCIENTISTS

9 March 1981

RECEIVED MAR 12 1981 U.S. NUCLEAR REGULATORY COMMISSION

Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Attention: Director, Division of Licensing

Dear People:

Re: Supplement to the Draft Environmental Statement (NUREG-0490) related to the operation of San Onofre Nuclear Generating Station, Units 2 and 3

Herewith are some brief comments on the above Supplement, in response to your invitation.

A-63

We are pleased that the NRC has finally published a document providing a hint of the consequences of severe accidents at the San Onofre Station. We consider, however, that this Supplement does not satisfy the intent of the Commission's Statement of Interim Policy of 13 June 1980 (Federal Register, 45, 40101). Nor does this Supplement provide the public with information sufficient to make a reasoned assessment of the risks of severe accidents at this plant.

You will recall that the Commission's Statement of Interim Policy followed a letter of 20 March 1980 from the Chairman of the Council on Environmental Quality (CEQ) to the Chairman of the NRC. Included in this letter was the statement:

> "The results of our review of impact statements prepared by the NRC for nuclear power reactors are very disturbing. The discussion in these statements of potential accidents and their environmental impacts was found to be largely perfunctory, remarkably standardized, and uninformative to the public."

This Supplement must be substantially revised and improved before it overcomes these CEQ criticisms. For guidance during this process of revision and improvement, the NRC staff should consult the report "NRC's Environmental Analysis of Nuclear Accidents: Is It Adequate?", prepared for CEQ by the Environmental Law Institute (ELI) in February 1980. A copy of this

1384 Massachusetts Avenue • Cambridge, Massachusetts 02238 • Telephone (617) 547-5552
1725 I Street, N.W. • Suite 601 • Washington, D.C. 20006 • Telephone (202) 296-5600

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report was provided to the NRC with the CEQ Chairman's letter.

Part 5 of the ELI report recommends that the NRC should continue, with some substantial improvements, its previous practice of studying a selection of accident scenarios. The ELI report recommends that this selection should be expanded to include "Class 9" accidents. Section 7 (Environmental Impact of Postulated Accidents) of the San Onofre Draft Environmental Statement (dated November 1978) exemplifies this previous practice; it estimates radiation doses for a number of selected accidents in Classes 1 through 8. This Supplement, however, merges nine release categories, weighted by assumed probabilities. The results of this analysis are confusing for the public; one might suspect that this is by intention.

Each accident scenario should be considered alone. For each scenario, the NRC should provide a clear account of:

(i) the nature of the postulated accident

(ii) the estimated nature of the radioactive release

(iii) the estimated nature of the environmental consequences of that release.

The Commission's Statement of Interim Policy directs:

> " . . . approximately equal attention shall be given to the probability of occurrence of releases and to the probability of occurrence of the environmental consequences of those releases."

This Supplement does not satisfy the intent of that directive. It merges these two probabilities although they are of quite different natures. One might suspect that this approach is selected in order to persuade the public that severe consequences have extremely low probabilities. This form of analysis and presentation does not fulfill the NRC's obligation to accurately inform the public.

As the NRC staff should well know, probabilities in nuclear accident analysis fall into two distinct categories:

(i) probability of occurrence of release
This category of probability concerns engineering estimates. These are very difficult to make since there is a limited statistical base and much of the uncertainty relates to human behaviour.

(ii) probability of occurrence of environmental consequences, given a particular release
This category of probability concerns factors such as wind speed and direction. These factors can be estimated from a good statistical base.

The NRC staff should revise this Supplement so as to exhibit their estimates of these probabilities separately, within each accident scenario studied.

The Commission's Statement of Interim Policy also directs:

> " . . . consequences shall be characterized in terms of potential radiological exposures to individuals, to population groups, and, where applicable, to biota."

This Supplement does not fulfill the intent of that directive. It provides very limited information on the geographical variation of potential exposure. More seriously, it provides essentially no information on the significance of exposure for different population groups. As the NRC staff should well know, certain population groups (especially children and fetuses) are at greater risk for a given release.

The importance of revising this Supplement, so as to accurately inform the public, can be illustrated by two estimates which can be gleaned from the supplement itself:

(i) probability of occurrence of the "PWR2" core melt accident
This release is one of the most severe accidents considered in the Reactor Safety Study (WASH-1400) and this Supplement. Table 7.1.4-2 of the Supplement estimates its probability as $7x10^{-6}$ per reactor-year. Section 7.1.4.2 concedes that this estimate could be low by a factor of 100. One thus finds (assuming a reactor life of 30 years) that this Supplement admits that a "PWR2" accident could have a 4% probability of occurrence during the life of San Onofre Units 2 and 3.

(ii) potential for serious health effects
Table 7.1.4-4 of this Supplement admits that a severe accident at San Onofre could lead to 130,000 acute fatalities, 300,000 subsequent fatal cancers, and 600,000 genetic effects.

In the light of the grave hazard shown by these estimates, the NRC has a clear duty to provide the public with more complete information than is contained in this Supplement.

Thank you for your attention.

Sincerely,

G.R. Thompson

Gordon Thompson, Ph.D.
Staff Scientist

GT:VN

UNITED STATES ENVIRONMENTAL PROTE[CTION AGENCY]
REGION IX
215 Fremont Street
San Francisco, Ca 94105

RECEIVED MAR 20 1981

16 MAR 1981

Project # DS-NRC-K06002-CA

Frank J. Miraglia, Acting Chief
Licensing Branch No. 3
Division of Licensing
Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Mr. Miraglia:

The Environmental Protection Agency (EPA) has received and reviewed the Draft Supplement (DS) to the Draft Environmental Impact Statement (DEIS) for the project titled SAN ONOFRE NUCLEAR GENERATING STATION, UNITS 2 AND 3.

In our previous reviews of environmental documents dealing with Light Water Reactors (LWR) EPA has consistently emphasized the need for a thorough evaluation of the environmental impacts from different LWR accident scenarios to include Class 9 accidents. The discussion of the environmental and societal impacts of a core melt down accident included in the Supplement to the Draft Environmental Impact Statement for the San Onofre Nuclear Generating Station, Units 2 and 3 is a step forward in this respect and, as a result, EPA applauds the Nuclear Regulatory Commission's (NRC) decision to prepare this Supplement.

The assessment of environmental impacts for severe accidents at the plant uses methodologies originally developed in the Reactor Safety Study (WASH-1460) and the Liquid Pathway Generic Study (NUREG-0440). Because these two studies will be the cornerstones for similar assessments for other nuclear power plants environmental statements, we would refer NRC to EPA's original technical comments on these studies. These comments can be found in "Reactor Safety Study (WASH-1400): A Review of the Final Report" and a letter from EPA's Office of Federal Activities to NRC dated February 8, 1977.

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Our specific comments on the San Onofre Supplemental DEIS and generic comments are attached. The EPA appreciates the opportunity to comment on this Draft Supplement. Should the NRC choose to revise other sections of the EIS, EPA would like to review these documents. If you have any questions regarding our comments, please contact Susan Sakaki, EIS Review Coordinator, at (415)556-7858.

Sincerely yours,

Jake Mackenzie, Director
Surveillance and Analysis Division

Attachment

EPA Technical Comments on the Supplement to the Draft Environmental Statement Related to the Operation of the San Onofre Generating Station Units 2 and 3 (NUREG-0490)

General Comments

The Final EIS for San Onofre Units 2 and 3 is dated March 1973. This statement contains a Section 7, titled "Environmental Impact of Postulated Accidents." It is not clear if the Supplement is to replace the original information or if the Supplement is supplemental. If this information is supplemental then we would suggest that the original Section 7 be revised to agree with the supplemental statements and data.

It would also be hoped that any previous information and conclusions would be revised if it is impacted by events occurring since 1973 or by a change in Commission consideration. For instance the supplement refers to the original Section 5.5 and further mentions 10 CFR Part 20 and 10 CFR Part 50. However, the supplement does not make any mention of the Commission's implementation of 40 CFR 190 for normal operation.

Specific Comments

Table 7.1.4-4

This table should correspond on a one-to-one basis with the release categories (PWR 1-9) in Table 7.1.4-2. It is also not readily apparent how the PWR 1-9 compares to the original Table 7.1.

Design Basis Accidents

In the discussion of accident risk and impact assessment of Design Basis Accidents (DBAs), Section 7.1.4.1, we do not understand the intent of the comparison of the results in Table 7.1.4-1 to the Reactor Site Criteria of 10 CFR 100. First, the infrequent accidents listed in Table 7.1.4-1 do not meet the requirements of 10 CFR 100 for purposes of site analysis. Footnotes to 10 CFR 100 state:

> (1)...calculations should be based upon a major accident, hypothesized for the purposes of site analysis...that would result in potential hazards not exceeded by those from any accident considered credible, and
>
> (2)...this 25 rem whole body value and the 300 rem thyroid value have been set forth as reference values, which can be used in the evaluation of reactor sites with respect to potential reactor accidents of exceedingly low probability of occurrence, and low risk of public exposure to radiation.

Secondly, by the description of infrequent accidents in the supplement ("events that might occur once during the lifetime of the plant"), these accidents have an annual probability of occurrences on the order of 10^{-2}, are considered credible, and are not of exceedingly low probability of occurrence. Reference to 10 CFR 100 and its implementation provide a misleading inference that, since the results shown in Table 7.1.4-1 are within the dose values of 10 CFR 100, the risk of those infrequent accidents is small and therefore acceptable. Also, the radiation doses listed in Table 7.1.4-1 are calculated using a conservative model approach which is relevant to safety evaluations and not consistent with the realistic approach to the assessment of environmental risks of normal operation and severe core melt accidents.

The discussion of impacts of infrequent accidents and limiting faults, in both the original DES and the Supplement, addresses probabilities of occurrence qualitatively. Yet, in the discussion of the more severe core melt accidents the probabilities of occurrence are quantified (Table 7.1.4-2). For consistency in the presentation of all environmental risks, the probabilities of occurrence of infrequent accidents and limiting faults DBA's should also be provided.

It is not clear whether the risks listed in Table 7.1.4-5, Annual Average Values of Environmental Risks Due to Accidents, include those from infrequent accidents and limiting faults (Table 7.1.4-2), postulated accidents (Table 7.2 of the original DES), and accidents leading to the PWR 1-9 release categories (Table 7.1.4-2). The risks should include all those from moderate frequency accidents, infrequent accidents, limiting faults and severe core melt accidents. Although the risk of the infrequent accidents and limiting faults is "judged to be extremely small" and appear to be overshadowed by the risk from core melt accidents, they should be fully presented. The risks from the more probable yet lower consequence accidents may indeed be significant to the individual risk and should be listed in the Supplement. It would also be beneficial to extend Figures 7.1.4-3, 7.1.4-5, and 7.1.4-7 to include the higher probability accidents.

It would be helpful to provide a summary table of the annual average value of environmental risks from operation of all the reactors at the San Onofre site. The risks

should include all those from normal operations, moderate frequency accidents, infrequent accidents, limiting faults and severe core melt accidents. Both societal and individual risks should be presented.

7.1.1.3 Health Effects

The statement that a dose greater than about 25 rem is necessary before any physiological effects to an individual are clinically detectable should be reviewed. Information contained in a World Health Organization technical report No. 123 would seem to indicate that physiological changes can occur at exposures as low as 10 rem.

7.1.3.3 Emergency Preparedness

It is unclear what is the basis of the statement, "Emergency preparedness plans including protective action measures for the San Onofre facility and environs are in an advanced, but not yet fully completed stage." The plans (seven) are at this date undergoing informal review by the Region IX Regional Assistance Committee (RAC). Thus, there has been no request for formal review, there has been no drill schedule established and there has been no full scale exercise. We do not concur in the Commission's statement that these plans are in an advanced stage.

Table 7.1.4-5

It is not clear from the information presented regarding risk and protective action that protective actions can be taken to reduce exposures by 10-20 times or in fact to prevent exposures determined by the State of California to be unacceptable considering the following:

1. The emergency preparedness plans and protective action measures for the San Onofre facility are not yet complete.
2. The State of California does not use the EPA's Protective Action Guides (PAG's).

In view of the above, we feel the statements made are premature.

Figure 7.1.4-8

This figure, "Relative Directional Risk to Individuals," might be a useful risk analysis. However, as presented, the figure is illegible and lacking in background information. It should be presented more clearly, with an accompanying table or coding explaining the significance of the numbers.

Decommissioning

The cost of reactor decommissioning and replacement power costs are as large as the costs from the Three Mile Island accident. It would seem that these costs could significantly change the cost-benefit information originally provided in Section 13. Future EIS's or Supplements to EIS's should include an evaluation of these costs.

San Diego
ASSOCIATION OF
GOVERNMENTS
Suite 524, Security Pacific Plaza
1200 Third Avenue
San Diego, California 92101
(714) 236-5300

March 19, 1981

Mr. Dino C. Scaletti
San Onofre Project Manager
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, DC 20555

Dear Mr. Scaletti:

A-68

On March 16, 1981, the Board of Directors of the San Diego Association of Governments (SANDAG) adopted a resolution supporting the operation of San Onofre Nuclear Power Plant Units 2 and 3 and requested the Nuclear Regulatory Commission to grant an operating license for these units subject to federal regulations regarding the safety of nuclear power plant operations and emergency planning for nuclear power plant accidents. This resolution and the supporting staff report are attached.

Please call me or have your staff call Steve Sachs of my staff if you have any questions about the Board of Directors action.

Sincerely,

for RICHARD J. HUFF
Executive Director

RJH/SS/sc

Attachments

cc: Patricia Fleming, SDG&E
Fred Massey, SCE

MEMBER AGENCIES: Cities of Carlsbad, Chula ...

San Diego ASSOCIATION OF GOVERNMENTS

RESOLUTION

No. 81-36

RESOLUTION SUPPORTING THE OPERATION
OF SAN ONOFRE NUCLEAR POWER PLANT
UNITS 2 AND 3
SUBJECT TO FEDERAL REGULATIONS REGARDING THE
SAFETY OF NUCLEAR POWER PLANT OPERATIONS AND
EMERGENCY PLANNING FOR NUCLEAR PLANT ACCIDENTS

WHEREAS, the Energy 2000 Task Force, appointed by Mayor Wilson of the City of San Diego, presented the conclusions and recommendations of its report to the SANDAG Board of Directors on February 23, 1981; and

WHEREAS, one of the recommendations of the Energy 2000 Task Force is to support the completion and operation of San Onofre Plants 2 and 3; and

WHEREAS, San Onofre Units 2 and 3, if completed and operated on schedule, will supply approximately half of the additional electricity needs forecast for the San Diego region between now and 1995; and

WHEREAS, the Nuclear Regulatory Commission will begin licensing hearings for San Onofre Units 2 and 3 in June 1981; and

WHEREAS, federal regulations concerning nuclear power plant safety and emergency response planning will have to be met in order for a license to be granted; NOW THEREFORE

BE IT RESOLVED that the Board of Directors supports the operation of San Onofre Nuclear Power Plant Units 2 and 3 and requests the Nuclear Regulatory Commission to grant an operating license for these units subject to federal regulations regarding the safety of nuclear power plant operations and emergency planning for nuclear plant accidents.

PASSED AND ADOPTED this 16th day of March 1981.

ATTEST: SECRETARY

CHAIRMAN

MEMBER AGENCIES: Cities of Carlsbad, Chula Vista, Coronado, Del Mar, El Cajon, Imperial Beach, La Mesa, Lemon Grove ...

SAN DIEGO ASSOCIATION OF GOVERNMENTS

RESOLUTION NO. 81-36 DATE CONSIDERED: 3/16/81

AGENCY	YES	NO	ABSENT	ABSTAIN
CARLSBAD	X			
CHULA VISTA	X			
CORONADO	X			
DEL MAR		X		
EL CAJON	X			
IMPERIAL BEACH	X			
LA MESA	X			
LEMON GROVE	X			
NATIONAL CITY	X			
OCEANSIDE	X			
SAN DIEGO	X			
SAN MARCOS	X			
SANTEE	X			
VISTA	X			
TOTALS	13	1		

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I certify from personal observation and count that the above results are an accurate record of the SANDAG Board of Directors vote and action.

Bettie Black

San Diego Association of Governments

BOARD OF DIRECTORS

DATE: March 16, 1981 AGENDA REPORT No.: R-95

CONSIDERATION OF SUPPORT FOR OPERATION OF SAN ONOFRE NUCLEAR POWER PLANT UNITS 2 AND 3

Introduction

The Board requested this report as the basis for considering a resolution to support the operation of San Onofre Nuclear Power Plant Units 2 and 3. Three important points the Board should consider before taking a position are:

- The risks to health and life of both present and future generations and the costs of reducing these risks associated with almost all aspects of the nuclear fuel cycle, are extremely controversial. There is little scientific or technical consensus on the severity of the risks and the effectiveness or cost of strategies to reduce these risks.
- San Onofre Units 2 and 3 would provide 440 MW of electric power to the San Diego region - almost one-half of the additional power requirements forecast to be needed between now and 1995 for the SDG&E Service Area by SDG&E and the California Energy Commission. These forecasts include the effects of existing conservation and alternative energy source programs which will reduce electricity demand. Potential additional electricity supplies and conservation and alternative energy sources which could result in a balance between demand and supply over the next 10 to 20 years without San Onofre Units 2 and 3 have been identified (see attachment for a partial list) but are not yet committed. In some cases, these sources may be infeasible or unavailable.
- The construction of San Onofre Units 2 and 3 is nearing completion. About one-half of the total $3.4 billion projected construction cost has been expended. The plant is currently undergoing U.S. Nuclear Regulatory Commission review in order to obtain an operating license.

It is my

RECOMMENDATION

that the Board of Directors support the operation of San Onofre Nuclear Power Plants 2 and 3 and request the Nuclear Regulatory Commission to grant an operating license for these units subject to federal regulations regarding the safety of nuclear power plant operations and emergency planning for nuclear plant accidents.

A-70

Discussion

San Onofre Units 2 and 3 are scheduled to have a total capacity of 2,200 megawatts (MW) of electricity. SDG&E is a 20% partner in the plant and is therefore entitled to 440 MW of the electricity generated. The other 1,760 MW is scheduled to be used by Southern California Edison Company (76%) and Municipal Utilities serving the Cities of Anaheim and Riverside (total of 4%).

The Nuclear Regulatory Commission (NRC) is the federal agency responsible for issuing nuclear power plant operating licenses. The NRC will hold hearings on the license applications for San Onofre Units 2 and 3 starting in June 1981.

There are many environmental and economic issues related to the operation of San Onofre Units 2 and 3 which include:

- Cost and reliability of nuclear power
- Risk of accidents from transport of uranium, spent nuclear fuel and operation of the plants.
- Cost of decommissioning the plants.
- Ability of the plants to withstand earthquakes.
- Hazards, cost and technical feasibility of long-term storage of radioactive wastes.
- Scope and adequacy of emergency plans to reduce radiation exposure in the event of an accident.

At the licensing hearings in June, it appears that the most controversial issues will be the ability of the plants to withstand earthquakes and the adequacy of emergency planning in case of an accident that could impact surrounding areas. The Plant must meet federal standards in both of these areas before a license will be issued.

RICHARD J. HUFF
Executive Director

San Diego ASSOCIATION OF GOVERNMENTS **RESOLUTION** No. 81-36

RESOLUTION SUPPORTING THE OPERATION
OF SAN ONOFRE NUCLEAR POWER PLANT
UNITS 2 AND 3
SUBJECT TO FEDERAL REGULATIONS REGARDING THE
SAFETY OF NUCLEAR POWER PLANT OPERATIONS AND
EMERGENCY PLANNING FOR NUCLEAR PLANT ACCIDENTS

WHEREAS, the Energy 2000 Task Force, appointed by Mayor Wilson of the City of San Diego, presented the conclusions and recommendations of its report to the SANDAG Board of Directors on February 23, 1981; and

WHEREAS, one of the recommendations of the Energy 2000 Task Force is to support the completion and operation of San Onofre Plants 2 and 3; and

WHEREAS, San Onofre Units 2 and 3, if completed and operated on schedule, will supply approximately half of the additional electricity needs forecast for the San Diego region between now and 1995; and

WHEREAS, the Nuclear Regulatory Commission will begin licensing hearings for San Onofre Units 2 and 3 in June 1981; and

WHEREAS, federal regulations concerning nuclear power plant safety and emergency response planning will have to be met in order for a license to be granted; NOW THEREFORE

BE IT RESOLVED that the Board of Directors supports the operation of San Onofre Nuclear Power Plant Units 2 and 3 and requests the Nuclear Regulatory Commission to grant an operating license for these units subject to federal regulations regarding the safety of nuclear power plant operations and emergency planning for nuclear plant accidents.

PASSED AND ADOPTED this 16th day of March 1981.

ATTEST: ______________________ SECRETARY ______________________ CHAIRMAN

MEMBER AGENCIES: ~~Cities of Carlsbad, Chula Vista, Coronado, Del Mar,~~ El Cajon, Imperial Beach, La Mesa, Lemon Grove, National City, Oceanside, San Diego, San Marcos, Santee and Vista/Ex-officio Member: California Department of Transportation/Honorary Member: Tijuana, B. C. A

ATTACHMENT
(From Energy 2000 Task Force Report)

Potential Supply Alternatives
For the SDG&E Service Area*
1980-2000

San Onofre 2 and 3	440 MW	(nuclear)
Arizona (renewed contract)	400 MW	(imported)
New Mexico (renewed contract)	150 MW	(imported)
Washington (renewed contract)	100 MW	(imported)
Mexico (purchase)	300 MW	(imported)
Geothermal	800 MW	(geothermal)
Blythe site	1,000 MW	(coal gasification)
Hydroelectric	34 MW	(hydroelectric)
Cogeneration	100 MW	(cogeneration)
Wind	30 MW	(wind)
TOTAL	3,354 MW	

SOURCE:

San Diego Gas and Electric Company, September 1979

*Some of these sources may be infeasible or unavailable. For example, Arizona Public Service Company would have to agree to a renewed contract for 400 MW of imported power from Arizona; the feasibility of 1000 megawatts from a coal gasification plant at Blythe has not been proved.

A-71

Southern California Edison Company

P. O. BOX 800
2244 WALNUT GROVE AVENUE
ROSEMEAD, CALIFORNIA 91770

K. P. BASKIN
MANAGER OF NUCLEAR ENGINEERING, SAFETY, AND LICENSING

TELEPHONE
(213) 572-1401

March 24, 1981

RECEIVED MAR 27 1981 U.S. NUCLEAR REGULATORY COMMISSION

Director, Office of Nuclear Reactor Regulation
Attention: Darrel G. Eisenhut, Director
Division of Licensing
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Sir:

Subject: Docket Nos. 50-361 and 50-362
San Onofre Nuclear Generating Station
Units 2 and 3

References: Realistic Estimates of the Consequences of Nuclear Accidents, M. Levenson and F. Rahn, EPRI, November, 1980.

This letter provides Southern California Edison Company's comments to the Supplement to Draft Environmental Statement related to the operation of San Onofre Nuclear Generating Station Units 2 and 3 NUREG-0490. In our review of this document we have found two points which we feel are in need of further clarification prior to the issuance of a Final Environmental Statement.

1. The following statement contained in Section 7.1.4.3,

> "The 200-rem whole-body dose figure corresponds approximately to a threshold value for which hospitalization would be indicated for the treatment of radiation injury. The 25-rem whole-body (which has been identified earlier as the lower limit for a clinically observable physiolgical effect) and 300-rem thyroid figures correspond to the Commission's guideline values for reactor siting in 10 CFR Part 100."

requires clarification, to prevent the statement from being misconstrued to state that San Onofre does not meet the Commission siting guidelines of 10 CFR 100.

In order to clearly differentiate between the Class 9 accident and the design basis accidents used in the Commission siting criteria, specific clarification is needed. The traditional Design Basis Accidents (DBA's) are hypothetical and conservative scenarios, evaluated in accordance with regulations and other regulatory guidance which define the required assumptions and methodology. In contrast, the Class 9 accident scenario is defined with no consideration of mitigation by engineered safety features, assumes highly conservative and consequence maximizing behavior of natural mitigation processes. Since the Class 9 accident uses much more conservative, unrealistic, assumptions, it is not considered in the evaluation of reactor siting.

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2. Although uncertainties in probability calculations are discussed in Sections 7.1.4.2 and 7.1.4.7 of the Supplement, the uncertainties in the source terms, and hence the consequences of the accident, are not discussed in either Section 7.1.4.3 or 7.1.4.7. These radiation source terms have been shown to be conservative by experiments performed at Rockwell, Karlsruke, Oak Ridge National Laboratory, General Electric (Aircraft Nuclear Propulsion Department), Bettis National Laboratory, Hanford National Laboratory, and tests performed in the Idaho Reactor Test Site. The results of these tests and experiments, summarized in a paper by M. Levenson and F. Rahn of the Electric Power Research Institute, indicate that natural processes are operating which prevent the release of radioactive nuclides from molten nuclear reactor fuel (Reference 1). Dr. Chauncey Starr, former President of the Electric Power Research Institute advised the Commission, at the Commissions November 18, 1980 meeting in Washington, D.C., that,

> "The important issue is that the initial review of this subject appears to indicate that under any conceivable realistic circumstance, the real source term is likely to result in risk to the public that is less by factors of 10 to 100 than that which was previously estimated."

Using Dr. Starr's estimate of a realistic maximum release into the atmosphere would lower the consequences (acute fatalities and cancer deaths) from a Class 9 accident by 1 to 2 orders of magnitude.

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The Final Environmental Statement for San Onofre Units 2 and 3 should be accurate, concise, and not leave room for misinterpretation. Where applicable, all sources of error, and the relative magnitude of error, should be indicated. We hope that these comments will help to make the FES for SONGS 2 and 3 such a document.

Very truly yours,

K P Baskin

APPENDIX B

NEPA POPULATION DOSE ASSESSMENT

CERTIFICATE OF SERVICE

I hereby certify that on July 20, 2020, I electronically filed the foregoing with the Clerk of the Court for the United States Court of Appeals for the Ninth Circuit by using the appellate CM/ECF system.

Participants in the case who are registered CM/ECF users will be served by the appellate CM/ECF system.

Dated: July 20, 2020

Respectfully submitted,
ALSTON & BIRD LLP

/ s/ Edward J. Casey

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No. 20-70899

**IN THE UNITED STATES COURT OF APPEALS
FOR THE NINTH CIRCUIT**

IN RE PUBLIC WATCHDOGS,

Petitioner,

v.

UNITED STATES NUCLEAR REGULATORY COMMISSION,

Respondent.

SOUTHERN CALIFORNIA EDISON COMPANY,
Intervenor.

INTERVENOR SOUTHERN CALIFORNIA EDISON'S SUPPLEMENTAL EXCERPTS OF RECORD

VOLUME 3 OF 8

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Attorneys for SOUTHERN CALIFORNIA EDISON COMPANY

INTERVENOR SOUTHERN CALIFORNIA EDISON'S SUPPLEMENTAL EXCERPTS OF RECORD: TABLE OF CONTENTS

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50	Division of Spent Fuel Management – Interim Staff Guidance – 2, Revision 2		3	SCE-SER-000801
44	NRC Safety Evaluation Report – Docket No. 72-1040 – HI-STORM UMAX Canister Storage System	Apr. 2, 2015	3	SCE-SER-000810
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67	NRC Supplemental Inspection Report	Jul. 9, 2019	4	SCE-SER-000864
55	NRC Inspection Report – San Onofre Nuclear Generating Station	Aug. 24, 2018	4	SCE-SER-000914
49	NRC Inspection of Independent Spent Fuel Storage Installation – Callaway Plant	Oct. 30, 2015	4	SCE-SER-00953
52	Final Safety Analysis Report on the HI-STORM UMAX Canister Storage System	Jun. 29, 2016	5 /6 / 7	SCE-SER-001053
57	NRC (Errata) San Onofre Nuclear Generating Station – Special Inspection Report	Dec. 19, 2018	7	SCE-SER-001710

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17	Division of Spent Fuel Storage and Transportation Interim Staff Guidance -1, Revision 2		7	SCE-SER-001768
36	Official Transcript of Proceedings – Nuclear Regulatory Commission – San Onofre Nuclear Generating Station Post-Shutdown Decommissioning Activities Report Hearing.	Oct. 27, 2014	7	SCE-SER-001779
60	NRC ISFSI Pad Surveys at SONGS		7	SCE-SER-001931
40	NUREG-1927 – Standard Review Plan for Renewal of Specific Licenses and Certificates of Compliance for Dry Storage of Spent Nuclear Fuel		8	SCE-SER-001935
13	IFST ISG-1, Rev.2, Division of Spent Fuel Storage and Transportation Interim Staff Guidance No. 1, Revision2, “Classifying the Condition of Spent Nuclear Fuel for Interim Storage and Transportation Based on Function”	May 11, 2007	8	SCE-SER-002060

Appendix B

NEPA POPULATION DOSE ASSESSMENT

Population dose commitments are calculated for all individuals living within 80 km (50 miles) of the facility employing the same models used for individual doses (see Regulatory Guide 1.109, in preparation). In addition, population doses associated with the export of food crops produced within the 80-km region and the atmospheric and hydrospheric transport of the more mobile effluent species such as noble gases, tritium, and carbon-14 have been considered.

B.1 NOBLE GAS EFFLUENTS

For locations within 80 km of the reactor facility, exposures to these effluents are calculated using the atmospheric dispersion models in Regulatory Guide 1.111 and the dose models described in Section 5.5 and Regulatory Guide 1.109. Beyond 80 km and until the effluent reaches the northeastern corner of the United States, it is assumed that all of the noble gases are dispersed uniformly in the lowest 1000 m (3280 ft) of the atmosphere. Decay in transit was also considered. Beyond this point, noble gases having a half-life greater than one year (e.g., Kr-85) were assumed to mix completely in the troposphere of the world with no removal mechanisms operating.

Transfer of tropospheric air between the northern and southern hemispheres, although inhibited by wind patterns in the equatorial region, is considered to yield a hemisphere average tropospheric residence time of about two years with respect to hemispheric mixing. Since this time constant is quite short with respect to the expected mid-point of plant life (15 years), mixing in both hemispheres can be assumed for evaluations over the life of the nuclear facility. This additional population dose commitment to the U.S. population was also evaluated.

B.2 IODINES AND PARTICULATES RELEASED TO THE ATMOSPHERE

Effluent nuclides in this category deposit onto the ground as the effluent moves downwind, which continuously reduces the concentration remaining in the plume. Within 80 km of the facility, the deposition model in Regulatory Guide 1.111 was used in conjunction with the dose models in Regulatory Guide 1.109. Site-specific data concerning production, transport, and consumption of foods within 80 km of the reactor were used. Beyond 80 km, the deposition model was extended until no effluent remained in the plume. Excess food not consumed within the 80-km distance was accounted for, and additional food production and consumption representative of the eastern half of the country was assumed. Doses obtained in this manner were then assumed to be received by the number of individuals living within the direction sector and distance described above. The population density in this sector is taken to be representative of the eastern United States, which is about 410 persons per km^2 (160 persons per mi^2). (This approach is conservative for San Onofre because population densities in the western United States are considerably lower than those in the eastern portion.)

B.3 CARBON-14 AND TRITIUM RELEASED TO THE ATMOSPHERE

Carbon-14 and tritium were assumed to disperse without deposition in the same manner as krypton-85 over land. However, they do interact with an atmospheric residence time of 4 to 6 years with the oceans being the major sink. From this, the equilibrium ratio of the carbon-14 to natural carbon in the atmosphere was determined. This same ratio was then assumed to exist in man so that carbon-14 to natural carbon in the atmosphere was determined. This same ratio was then assumed to exist in man so that the dose received by the entire population of the United States could be estimated. Tritium was assumed to mix uniformly in the world's hydrosphere, which was assumed to include all the water in the atmosphere and in the upper 70 m (230 ft) of the oceans. With the model, the equilibrium ratio of tritium to hydrogen in the environment can be calculated. The same ratio was assumed to exist in man, and was used to calculate the population dose, in the same manner as with carbon-14.

B.4 LIQUID EFFLUENTS

Concentrations of effluents in the receiving water within 80 km of the facility were calculated in the same manner as described above for the Appendix I calculations. No depletion of the nuclides present in the receiving water by deposition on the bottom of the Pacific Ocean was assumed. It was also assumed that aquatic biota concentrate radioactivity in the same manner as was assumed for the Appendix I

evaluation. However, food consumption values appropriate for the average individual, rather than for the maximum, were used. It was assumed that all of the sport and commercial fish and shellfish caught within the 80-km area were eaten by the U.S. population.

Beyond 80 km, it was assumed that all of the liquid effluent nuclides except tritium have deposited on the sediments so they make no further contribution to population exposures. The tritium was assumed to mix uniformly in the world's hydrosphere and to result in an exposure to the U.S. population in the same manner as discussed for tritium in gaseous effluents.

APPENDIX C

EXPLANATION AND REFERENCES FOR BENEFIT-COST SUMMARY

Appendix C

EXPLANATION AND REFERENCES FOR BENEFIT-COST SUMMARY

C.1 ECONOMIC IMPACT OF STATION OPERATION

C.1.1 Direct benefits

C.1.1.1 Energy

2114 MWe x 1000 kW/MW x 365 days x 24 hr/day x capacity factor (0.5 or 0.7). This product ranges from 9.3 x 10^9 kWhr/year (0.5 capacity factor) to 13.0 x 10^9 kWhr/year (0.7 capacity factor).

C.1.1.2 Reduced regional oil consumption

Section 8.3.1 shows that the applicants primarily have oil/gas fired units, which would have to be operated to a greater extent if SONGS 2 & 3 are not operated. The additional fuel oil consumption (assuming a 50% capacity factor for the nuclear units) is calculated as follows:

$$\frac{9.3 \times 10^9 \text{ kWhr} \cdot 9{,}000 \text{ Btu/kWhr} \cdot 1 \text{ bbl oil}}{6.29 \times 10^6 \text{ Btu}} = 13.2 \times 10^6 \text{ bbl oil.}$$

C.1.2 Economic costs

C.1.2.1 Fuel

From Sect. 8.3.1, the staff's estimate of fuel cost is $10.8 per megawatt-hour in 1983. Assuming a 60% capacity factor or 11.1 x 10^6 MWhr/yr gives the value in Table 10.1.

C.1.2.2 Operating and maintenance

Using the staff's OMCST computer code, operating and maintenance costs are estimated to be 4.05 mills/kWhr at 60% capacity, which multiplied by 11.1 x 10^9 kWhr/year gives the values in Table 10.1.

Decommissioning: Based on estimates given in Sect. 9.4, the cost of decommissioning each unit will be $66.7 million in 1978 dollars or $85.4 million in 1980 dollars at the end of the useful life of the plant. If this value is discounted from 2013 to 1983, then annualized over a 30-year life assuming a real interest and discount rate of 4.76%, and then multiplied by 2 units, the value in Table 10.1 is obtained.

APPENDIX D

CULTURAL RESOURCES

STATE OF CALIFORNIA—THE RESOURCES AGENCY EDMUND G. BROWN JR., Governor

DEPARTMENT OF PARKS AND RECREATION
P.O. BOX 2390
SACRAMENTO 95811
(916) 445-8006

DEC 18 1980

Mr. Dino Scaletti
Environmental Projects
Division of Site, Safety,
and Environmental Analysis
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Mr. Scaletti:

San Onofre Nuclear Generating Station,
Units #2 and #3, Operating License Stage

My staff has recently completed review of the "National Register Assessment Program of Cultural Resources of the 230 KV Transmission Line Rights-of-Way from San Onofre Nuclear Generating Station to Black Star Canyon and Santiago Substation and to Encina and Mission Valley Substation", prepared by WESTEC Services, dated September 1980.

In accordance with the provisions of the Advisory Council on Historic Preservation's Procedures set forth in 36 CFR 800, Section 106 of the National Historic Preservation Act of 1966 and the Memoranda of Agreement of October 29, 1979, I have the following comments to offer:

1. Based on the information I have been provided, I concur that the following sites are not eligible for National Register of Historic Places: CA-Ora-419, Ora-823, Ora-786, Ora-787, Ora-700, Ora-782, Ora-784, Ora-785, Ora-832, SDi-6693, SDi-6131, SDi-5444, SDi-6136, SDi-6137, SDi-6150, SDi-6151, and SDi-6152.

2. Sites CA-Ora-640, Ora-458, and SDi-6133 are outside the area of potential environmental impact for this undertaking.

3. I do not concur that site CA-Ora-824 is not eligible for the National Register of Historic Places. I feel that this site may be eligible based on Bean and Vane's findings in 1979 that this site possesses a high potential for significance.

4. I concur that the following sites are eligible for inclusion in the National Register as important components of the proposed San Joaquin Archeological District: CA-Ora-495, Ora-496, and Ora-499.

5. The following sites have been determined eligible for inclusion in the National Register as important components of the Upper Aliso Creek Archeological District: CA-Ora-447, Ora-438, and Ora-725.

6. The following sites should also be included as eligible properties within the Upper Aliso Creek Archeological District: CA-Ora-905, Ora-828, Ora-825, Ora-826, and Ora-827.

7. I concur that the following sites are eligible for inclusion in the National Register as significant components of the proposed Santiago Creek Archeological District: CA-Ora-829, Ora-830, and Ora-831.

8. I concur that the following sites are eligible for inclusion in the National Register as significant components of the proposed Agua Hedionda Archeological District: CA-SDi-6135, SDi-6133, and SDi-6140.

9. I also concur that the following sites are locally significant and are eligible for the National Register under Criterion "d" (36 CFR 1202.6): CA-Ora-498, SDi-4538, SDi-6130, SDi-6138, and SDi-6149.

10. Formal determinations of eligibility for these sites and districts should be sought from the Keeper of the Register in accordance with 36 CFR 1204.

11. I concur with the report's findings that this undertaking will have No Effect on eligible sites CA-Ora-905, Ora-828, Ora-826, Ora-827, Ora-829, and SDi-4538.

12. I concur with the report's findings that operation and maintenance (O&M) of access roads will affect the following eligible sites: CA-Ora-498, Ora-824, Ora-495, Ora-447, Ora-496, Ora-499, Ora-825, Ora-725, Ora-830, Ora-831, and SDi-6130. However, I feel that there will no No Adverse Effect on these resources if one of the two following conditions can be met:

    a. Access roads can be covered with a chemically inert, visually distinguishable fill within the boundaries of these sites in a manner which will preclude future ground disturbance of the cultural deposit during future O&M activities on access roads, or;

    b. O&M activities can be restricted to access roads, and the remaining research potential of surface artifacts within the provenience of existing access roads can be used to define the important factors which should be considered in determining the effects of continued disturbances as proposed in the Cultural Resource Management Plan on page 359 of the subject report. This program should be oriented towards defining the value of research potential and the effects that various activities may have on disturbed surface sites in similar environmental contexts. The program should also be responsive to the Advisory Council's Supplementary Guidance for Treatment of Archeological Properties supporting a No Adverse Effect Determination.

13. The information I have been provided indicates that undisturbed cultural deposits will be affected by O&M of access roads in the vicinity of site CA-Ora-438. However, it is my opinion that there will be No Adverse Effect if one of the two following conditions can be met:

    a. Access roads can be covered with a chemically inert, visually distinguishable fill within the boundaries of this site in a manner which will preclude future ground disturbance of the cultural deposit during future O&M activities, or;

    b. O&M activities can be restricted to access roads, and a Data Recovery Plan is implemented in accordance with the Advisory Council's Supplementary Guidance for Treatment of Archeological Properties supporting a No Adverse Effect Determination. The rationale for this recommendation is stated in the above referenced Guidance on pages 10 and 11, "An Undertaking may be taken to have no adverse effect...if the agency is committed to a data recovery program...if...the property is shown to be subject to destruction and deterioration regardless of the undertaking, so the agency's action is only slightly hastening a process that is inevitable in any event."

14. O&M activities and construction will have an effect on sites CA-SDi-6135, SDi-6138, SDi-6149, and SDi-6140. However, it is my opinion that there will be No Adverse Effect on these sites if a Data Recovery Plan is implemented in accordance with the Advisory Council's Supplementary Guidance for Treatment of Archeological Properties supporting a No Adverse Effect Determination. The rationale for this recommendation is the same as that cited in Item 13.b. above.

15. Concurrence of these determinations of effect should be sought from the Advisory Council in accordance with 36 CFR 800.4.c.

If you should have any questions, please contact Daniel Bell of my staff at (916) 322-8702.

Sincerely,

K Mellon

Dr. Knox Mellon
State Historic Preservation Officer
Office of Historic Preservation

D-6317D

cc: Mr. L. Jack Brunton
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Mr. Charles Niquette
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Lake Plaza-South, Suite 616
44 Union Boulevard
Lakewood, CO 80228

APPENDIX E

CALIFORNIA COASTAL COMMISSION,
MARINE REVIEW COMMITTEE REPORT

CALIFORNIA COASTAL COMMISSION
631 Howard Street, San Francisco 94105 — (415) 543-8555

TO: State Commissioners

FROM: Michael Fischer, Executive Director

SUBJECT: Report of San Onofre Nuclear Power Plant Marine Review Committee (For Commission consideration at the February 17-19 Meeting.)

Summary

The 1974 permit for the San Onofre Nuclear Power Plant's Units 2 and 3 established a three member Marine Review Committee (MRC) to study the effects of the Plant's cooling system on ocean life and to make recommendations to the Commission. Units 2 and 3 of the Plant are not yet operational. The MRC has submitted a report (conclusions attached) predicting effects on fish, kelp, plankton and other ocean life. The MRC recommends against any design changes to the cooling sustem at this time. Staff recommends the Commission take note of the MRC recommendations and endorse a future monitoring program to determine actual effects on ocean life in the future after system operation. If substantial adverse effects are found, the Commission can impose design or operational changes or mitigation measures, based on MRC recommendations. But, given MRC predictions, major system design changes in the future seem unlikely.

E-1

Background

The Commission's predecessor Coastal Zone Commission approved the construction of Units 2 and 3 of the San Onofre Nuclear Generating Station (SONGS) on February 20, 1974 (Permit No. 183-73). Condition B of the Permit provided for the establishment of an applicant funded Marine Review Committee (MRC) composed of an appointee of the State Commission, an appointee of Southern California Edison Company, and an appointee of the appellants. The appellants are coordinated by Friends of the Earth. The Condition provides for the MRC to undertake a "comprehensive and continuing study of the marine environment offshore from San Onofre...to predict, and later to measure, the effects of San Onofre Units 2 and 3 on the marine environment..." (Condition B1).

The MRC can make recommendations to the Commission, based on MRC studies, and the recommendations can include changes that the MRC believes necessary in the cooling system for Units 2 and 3. This cooling system takes in large amounts of seawater to cool the units and then discharges the heated water back to the ocean. Condition B6 of the Permit states:

> Should the study at any time indicate that the project will not comply with the regulatory requirements of State or Federal water quality agencies, or that substantial adverse effects on the marine environment are likely to occur, or are occurring, through the operation of Units 1, 2, and 3, the applicants shall immediately undertake such modifications to the cooling system as may reasonably be required to reduce such effects or comply with such regulatory requirements (which can be made while construction is going on and could be as extensive as requiring cooling towers if that is the recommendation). The State Commission shall then further condition the permit accordingly.

Thus, the Commission can impose new conditions on the cooling system only if the conditions are based on MRC recommendations and the Commission judges the conditions to be "reasonable". New conditions can be based only on an MRC finding that "substantial adverse effects on the marine environment are likely to occur, or are occuring, through the operation on Units 1, 2, and 3...."

Since its beginning, the MRC has submitted a number of reports to the Commission. After receiving an MRC report in mid-1979 the Commission, at its November 21, 1979 meeting, asked the MRC to take one final "best shot" at predicting effects on the marine environment prior to the start of Nuclear Regulatory Commission (NRC) hearings on the operating license for Units 2 and 3. The MRC has now submitted that report, MRC Document 80-04(I). The conclusions are attached to this staff report, and the MRC will present the conclusions to the Commission at its January 20-22 meeting.

Staff Analysis

The Marine Review Committee has, over the last six years, conducted monitoring and predicting studies that seem to be as comprehensive and thorough as possible given the state-of-the-art in predicting effects on the large and dynamic nearshore ocean environment. It is possible that the square kilometer offshore SONGS is the most heavily sampled and studied patch of the ocean anywhere. Predicting the effects of the SONGS cooling system on ocean life has had to face a number of inherent difficulties, including: understanding the life cycles of ocean organisms; obtaining enough samples over a long enough time period to enable statistical analyses; developing quantitative models of water flows, turbidity and population dynamics; and, most important, attempting to separate out effects or likely effects of the cooling system from other major factors affecting ocean life, including storms, water temperature and chemistry changes, fishing, changes in nutrient levels, changes in migratory habits, and natural population fluctuations.

Design Changes. The MRC has needed to use models and numerous assumptions in assessing possible effects on living ocean populations. Such exercises can give scenarios, but not high confidence predictions. The MRC report consequently presents a number of estimates of future effects on fish larvae, small shrimp, plankton, and and a kelp bed. It does not, however, state that these effects are likely or certain to occur, and, therefore, it does not state that "substantial adverse effects on the marine enviroment are likely to occur", as required in Condition B6 for modification of the cooling system. The report, then, explicitly recommends against design changes in the cooling system at this time, while stating "it is possible that we have grossly underestimated the ecological consequences of SONGS Units 1, 2, and 3" (Page 7). The actual effects can only be determined through monitoring the ocean environment after the Units become operational. The MRC has extensive results from pre-operational sampling and data collection and will be in a position to implement a useful post-operational monitoring program. Staff is therefore recommending the Commission endorse a continued MRC monitoring program and ask that the program design and budget be submitted to the Commission. If the MRC finds "substantial adverse effects" the Commission may still impose conditions accordingly.

Mitigation. One such condition could involve mitigation for damage determined by the MRC. The Commission directed the MRC to explore mitigation alternatives. This last attempt at predictions has taken up most MRC time, and the MRC report states it will recommend to the Commission which mitigation measures, in addition to artificial reefs for kelp, should be examined.

Radiological Monitoring. A 1979 MRC report detailed a number of inadequacies in the radiological monitoring program in the ocean around SONGS. The Commission directed staff to report these inadequacies to the Southern California Edison Co., the Nuclear Regulatory Commission, and the California Department of Health Services and to pursue remedies. SCE has since revised its radiological monitoring program extensively and has submitted it to the NRC. Both the NRC and the MRC author of the previous report are evaluating the revised program at present.

Staff Recommendation

Staff recommends the Commission adopt the following resolution:

The Commission thanks the Marine Review Committee for the report "Predictions of the Effects of San Onofre Nuclear Generating Station and Recommendations", adopted unanimously by the members of the MRC. The Commission notes that the MRC does not predict at this time that substantial adverse effects on the marine environment are likely to occur fron the operations of the SONGS cooling system, and that the MRC recommends against system design changes at this time. However, the Commission also notes that the MRC states it may have grossly underestimated these effects. The Commission agrees, therefore, that the MRC should conduct a comprehensive and thorough monitoring program of the effects after SONGS becomes operational and requests that the MRC submit the design and cost of such a program to the Commission. If such monitoring discovers substantial adverse effects on the marine environment, the Commission can, at that time, based on MRC recommendations, impose new conditions including design or operating changes or mitigation measures. The Commission recognizes, given the MRC predicted effects of the cooling system, that future imposition of any major design changes to the cooling system is unlikely.

marine review committee

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DEPT. OF BIOLOGICAL SCIENCES
UNIVERSITY OF CALIFORNIA
SANTA BARBARA, CA 93106

November 17, 1980

RECEIVED
NOV 24 1980
CALIFORNIA
COASTAL COMMISSION

Mr. Bill Ahearn
California Coastal Commission
4th Floor
631 Howard Street
San Francisco, California 94105

Dear Bill:

This letter formally transmits to the California Coastal Commission, under separate cover, the Marine Review Committee's predictions concerning the effects of San Onofre Units 1, 2 and 3 upon the marine ecosystem. The Report also contains a study of options and a set of recommendations to the Commission. These predictions and recommendations have been agreed upon unanimously by the Committee. The Appendices will follow in approximately two weeks.

A later report will discuss mitigation in more detail.

Yours sincerely,

Rimmon C. Fay

Rimmon C. Fay

Byron Mechalas

Byron Mechalas

W. Murdoch

William W. Murdoch
(Chairman)

REPORT OF THE MARINE REVIEW COMMITTEE TO THE CALIFORNIA COASTAL COMMISSION: PREDICTIONS OF THE EFFECTS OF SAN ONOFRE NUCLEAR GENERATING STATION AND RECOMMENDATIONS
PART I: RECOMMENDATIONS, PREDICTIONS, AND RATIONALE

Marine Review Committee

William W. Murdoch, Chairman
University of California

Byron J. Mechalas
Southern California Edison Company

Rimmon C. Fay
Pacific Bio-Marine Labs, Inc.

MRC Document 80-04(I)

CONTENTS

INTRODUCTION

The Marine Review Committee was charged, in Permit No. 183-73 of the California Coastal Commission, to carry out "a comprehensive and continuing study of the marine environment offshore from San Onofre . . . to predict, and later to measure, the effects of San Onofre Units 2 and 3 on the marine environment, . . . in a manner that will result in the broadest possible consideration of the effects of Units 1, 2 and 3 on the entire marine environment in the vicinity of San Onofre." This Report responds to the charge to predict the effects of Units 2 and 3.

San Onofre Nuclear Generating Station (SONGS) Unit 1 has been operating since 1968. Almost 150 billion gallons of seawater per year circulate through the Plant. Water flows in through a single intake and is discharged through a single discharge pipe at 19°F above the intake temperature. The construction of SONGS Units 2 and 3 is virtually completed. Each has a single intake, each drawing in seawater at a rate of 830,000 gallons per minute, which will result in an estimated flow of almost 700 billion gallons per year. Each also discharges its heated effluent through a series of 63 diffuser ports set along a kilometer-long pipe that tapers from 18' to 10'-14' in diameter (Figure 1, Maps 1 and 2). This discharged water moves rapidly towards the surface, entraining and moving with it roughly 10 times its own volume of water. As it spreads, this water mass moves various distances offshore, depending upon the prevailing currents. MRC has measured these currents, and Southern California Edison has produced a physical model of SONGS' water movement.

The effects of the cooling system of Unit 1 upon the marine ecosystem were described in MRC Annual Reports for 1978 and 1979. The documented effects are restricted to a region within a kilometer or two of SONGS. In seeking to predict the effects of Units 2 and 3, MRC has looked at the loss of organisms taken into the intakes, the possible losses caused by water movements driven by the diffuser plumes, and the effects of the diffusers and heat treatments on the physical environment, and hence upon the biota.

The predictions presented in this Report are in most cases close to final. Although we can and will obtain some more information on the major parts of the ecosystem near SONGS before Units 2 and 3 begin operation, we have obtained most of the information it is possible to obtain with a feasible expenditure of effort. Where major uncertainties remain, further study will not in general resolve them; they are largely an inescapable result of the practical difficulties in studying real ecological systems, and of the nature of such systems. The exceptions are kelp, where future work should provide more, and important, information, and some modelling studies that have not yet been completed. At this point, however, future work on predictions is aimed mainly at guiding our monitoring studies.

Following this Introduction, the Report presents our recommendations. There follows a brief statement of predictions for each major part of the community, and a more extensive Rationale, which explains how we arrived at the predictions. The Rationale unavoidably contains some technical discussion, but we have tried to write it so that the reader unfamiliar with the study can follow it. Finally, a series of separate Appendices accompanies this Report. These appendices are the reports of various contractors, and

analyses (by MRC and its consultants) of a number of difficult technical issues. The Rationale refers to those Appendices, where necessary, by project, number and, if appropriate, page number.

We would like to stress two findings that have general importance for management of and planning for nearshore coastal waters in California. First, we reiterate a previous conclusion that, in open coastal situations, a diffuser design is likely to be ecologically more damaging than a single point discharge, even though the latter would violate present State thermal discharge standards.

Second, we have recently obtained evidence that the early (larval) stages of nearshore sport and commercial fish species (e.g. bass, halibut) are particularly sparse very close to shore, while the larvae of fodder fish species are abundant right into shallow waters. Fodder fish populations are probably better able than sport and commercial species to withstand additional mortality on their larval stages. If this pattern holds along the whole California coast, it should be used as basic information in future planning - e.g. the placement of intakes and outfalls. This is not a blanket recommendation for placing structures close to shore, but rather a recommendation to weigh the possible losses of fish larvae in such decisions.

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OPTIONS AND RECOMMENDATIONS

Options

San Onofre Kelp bed (SOK) and nearshore fish populations are the major parts of the marine ecosystem that SONGS Units 1, 2 and 3 could significantly harm. Mysids, and perhaps zooplankton, are of less direct interest to society, but they also might sustain significant and quite large impacts. In the light of the predictions, MRC reviewed a number of possible recommendations that could be made to the Commission:

1. Make no design changes at this time. Monitor the effects.

2. Make no design changes at this time. Examine the feasibility of mitigating some or all of the effects, with a view to recommending mitigation measures to the Commission.

3. Extend the intake pipes to beyond the 30 meter depth.

4. Redesign the diffusers of Units 2 and 3, to convert them to single point discharges, located either 4 to 5 km offshore or very close inshore.

5. Convert the once-through cooling system to cooling towers.

Option 1 would require only a monitoring program, which would be carried out over several years to determine the effects of SONGS on the marine ecosystem. This program, in addition, would generate important information for future coastal planning, and would test how well we can predict the ecological consequences of a major coastal installation.

Option 2 MRC has completed a short "paper" feasibility study of certain kinds of mitigation (Mitigation Appendix). This study describes various methods of enhancing the production of economically important species, such as reef fish and abalone. Southern California Edison has

established an experimental reef aimed at producing a kelp bed and associated organisms, including fish and abalone. Other mitigation measures may be feasible.

It should be stressed that mitigation could not be expected to replace completely the biota lost through SONGS' operation. San Onofre Kelp bed could perhaps be replaced by a similar kelp bed, but fish losses would probably be replaced (partially) by a somewhat different mix of species. Lost mysids and plankton are not likely to be replaced by any known mitigation measure. An adequate mitigation study would therefore need to address the acceptability of "replacing" losses of one species by increasing the production of another.

Option 3 The possibility of extending the intakes out to deeper water was suggested previously (MRC 1979 Interim Report) as a means of (1) reducing the turbidity of intake water, so that the effects on SOK would be reduced, and (2) reducing the kill of nearshore fish larvae. With regard to aim (1), the turbidity study (Turbidity Appendix) suggests that much of the turbid water passing over SOK will originate at the inshore segment of the diffusers and will be carried offshore by secondary entrainment, so that the gain from changing the intakes would be relatively small. With regard to aim (2), our recent analyses show that the larvae of nearshore sport and commercial species are relatively sparse in the present intake area, and are quite dense out to about 7 km offshore. The gain in moving the intakes offshore would therefore be mainly a reduction in fodder fish kills, while we would likely kill more of sport and commercial species.

Option 4 The diffusers carry turbid water over the kelp bed. They

also will cause an unknown, but probably significant, amount of mortality in mysids, plankton and fish larvae. A single point discharge would greatly reduce this latter mortality, and moving the discharge either close inshore or further offshore would remove the kelp bed from the influence of the discharge. A single point discharge would violate the State thermal tolerances, but MRC believes this would cause much less ecological damage than the diffusers. It might be possible to make practical use of the waste heat from an inshore discharge. MRC has not evaluated in detail the ecological consequences of these two alternatives.

Recommendations

We recommend Options 1 and 2, and recommend against design changes at this time (Options 3, 4 and 5).

Monitoring is needed to measure the effects of Units 1, 2 and 3, as required by the Permit. It is also essential that the effects are measured and compared with MRC's quantitative predictions. Part of our study is a unique effort to make such predictions, and it is only by testing them that we can determine if such prediction is possible, how accurate it is, and what changes are needed to make better predictions in future planning. Predictions of probable effects, whether made explicit or not, are of course an integral part of all coastal planning.

We also recommend that MRC's remaining and ongoing prediction efforts be completed. These are now small studies. Such quantitative predictions are important, not only in themselves, but as a guide to the future monitoring program.

It is important to monitor the success of Southern California Edison's experimental reef, now established some 5 km south of SONGS. The evidence

on the efficacy of reefs, especially as a basis for new kelp beds, is equivocal and in contention, and this experiment will allow us to judge the best available California reef technology. MRC will present to the Commission, at a later date, a recommendation on whether or not other mitigation measures should be examined.

We recommend against moving the intake pipes (Option 3), for the reasons given under that Option. We also recommend against Options 4 and 5 at this time. Destruction of the offshore portion of the kelp bed is a major possible effect of the diffusers. However, at this moment we are not certain this will occur, and it is also possible that the effect could be mitigated. Some mitigation of fish losses may also be possible.

It is possible that we have grossly underestimated the ecological consequences of SONGS Units 1, 2 and 3. If monitoring proves this to be the case, we will re-examine the possibility of recommending major design changes.

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PREDICTIONS

FISH

Introduction

Most fish caught in Southern California are netted by commercial fishermen, and most come from fishing areas more than a few kilometers off the coast. By contrast, most sport fish in Southern California are caught close to the land - within the 33 California Fish and Game "fishing blocks" that are contiguous with the shore. In this Report we are concerned mainly with those sport fish and with commercial catches taken close to shore, for it is only this nearshore group of fish that SONGS is expected to affect. In evaluating the predictions, therefore, it should be kept in mind that SONGS is not expected to influence the great bulk of the fish populations that are harvested by California fishermen.

The species that concern us are fish that live as adults mainly within about 4 or 5 km of shore and that produce planktonic (drifting) eggs and larvae in the same zone. Among these species there are two groups: the nearshore sport and commercial species, the harvest of which is made up mainly by halibut, white seabass, kelp bass and sand bass, and the nearshore fodder fish (or forage fish) that form a major portion of the prey of the sport and commercial species.

In the predictions, we present various numbers to help the reader evaluate the likely effects of SONGS. It is easy to misinterpret these numbers, and we give here some essential background information. If we know the abundance and sizes of all of the halibut, say, in some area along the

coast, we can calculate the total living weight (biomass) of halibut in that region. This is called the standing stock. Each year, there are additions to this standing stock - some individuals that were larvae grow up to become adults, and many of those already adult grow and gain weight. If we could add up all the accumulated growth (in weight) we would be able to say how much new biomass had been added to the population. This is the annual production of new halibut tissue. We cannot estimate this directly, but a general rule of thumb is that a sport and commercial population gains about 60% of its standing stock weight per year. If our harvesting techniques were perfect we could take all of this production each year as harvest, and keep the standing stock steady from one year to the next. However, inevitably some fish die of disease and parasites, others are eaten by predators, and so on. The annual harvest, therefore, is always less than the annual production. In these nearshore sport and commercial species near San Onofre we estimate the harvest is roughly a quarter of production.

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As long as the harvest plus other factors do not take more than the annual production, the population will not decline. However, if, on average, harvest plus other losses are greater than production, the population will decline. If they are less than production, the population will increase, until it approaches a limit (say its food supply), at which time production will begin to decline and the population will level off.

We stress that the numbers given below are in all cases approximate. They give us an indication of the likely size of effects, but they do not tell us precisely what losses will be.

Predictions

1. Nearshore Sport and Commercial Fish

It is probable that, because of SONGS' activities, somewhere between 27 and 60 tons of nearshore sport and commercial fish production will be lost annually (Table 1). We feel the lower figure is more probable than the upper figure. Halibut is the species that will be most affected. Fish move about, so any loss of production will be spread over some area. We do not know how large an area, and provide a comparison between the consequences of spreading the loss over a small (45 km) and a large (300 km) stretch of coastal waters.

A loss of 27 tons would be equivalent to about 6% of the annual production of nearshore sport and commercial fish in the four fish blocks covering about 45 km of coastline near SONGS. It is equivalent to about one-third of the most recently documented (1975) harvest of these species from these four fishing blocks (85 tons). This does not mean that all of the losses will occur in these four blocks, or that the harvest can be expected to decline by either 6% or one-third.

If the losses were to be spread evenly over 300 km (about three-quarters of the length of the California Bight), then the loss in annual production over this area would be 1%. The loss in harvest could be more than 1% of that caught over 300 km. For example, to take an extreme case, if all natural losses are unavoidable, then all of the loss would come out of the harvest, which, for the 1975 harvest, would decline by roughly 10%.

There is quite strong evidence that the stocks of nearshore sport and commercial fish (especially halibut) have declined in the past two decades. We believe that these populations are unlikely to be able to compensate for

(i.e. make up for) significant additional mortality. However, the projected loss of sport and commercial fish, caused by SONGS, is sufficiently small that we believe it will not, in itself, have a significant effect on these populations.

Although SONGS alone is expected to have a minor effect upon the populations of nearshore sport and commercial fish, the cumulative effect of a number of sources of mortality of this order would be expected to contribute to continued decline in these populations. Future planning in the California Bight, therefore, should not evaluate additional installations and other environmental insults as independent events, but should consider their cumulative effects.

2. Fodder Fish

Anchovies probably contribute more than any other species to the diet of nearshore sport and commercial fish. Although enormous numbers of anchovy larvae will be killed by SONGS, we do not expect this vast population to be affected as a result of the operation of SONGS.

Nearshore fodder fish species are also important in the diets of nearshore sport and commercial fish. The two most abundant nearshore fodder fish are queenfish and white croaker. SONGS is expected to cause a loss in production of nearshore fodder fish of at least 300 tons per year.* Unlike the sport and commercial species, there is no evidence that the fodder fish populations are declining, so that we could expect some compensation for these losses. We do not know how much, so we cannot predict a precise net loss. Fodder fish in general move around more than sport and commercial species, and the populations in the entire Bight may well be thoroughly

*All weight figures are wet weight and are in metric tons.

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mixed, so that losses would be spread over the Bight (roughly 400 km). If the losses were spread over the Bight, and if no compensation occurred, they would be equivalent to about 7% of the annual production of these fish.

The projected loss of the equivalent of 300 tons of fodder fish production is owing mainly to the loss of larvae in the intakes. We expect there will be additional losses caused by the diffusers carrying larvae to inhospitable environments offshore. These losses could be very large - greater than those caused by the intakes - but we cannot predict them accurately.

The projected intake losses alone are sizeable. While we cannot estimate how the populations will be affected (because we do not know enough about compensation), the accumulation of effects of this order would be expected eventually to cause declines in these stocks. Thus, while SONGS itself may not cause such declines (and we do not know whether it will or not), we would be concerned about accumulating additional losses of this magnitude in the future.

We expect that the direct impingement of juvenile and adult fodder fish (mainly queenfish) in the intakes will cause measurable changes in the age structure and sex ratio of this species to a distance of several kilometers from SONGS.

3. Mechanisms

Fish losses are caused by three main mechanisms: (1) direct impingement of juvenile and adult fish in the intakes, (2) loss of immature stages (especially larvae) in the intakes, and (3) loss of immature stages in the diffusers. Mechanisms (2) and (3) are the most important. The diffusers could kill larvae (a) through subjecting them to turbulent shear and (b) by

carrying inshore larvae to an inhospitable environment offshore (translocation).

Intake losses: Our recent analyses have yielded a critical piece of information that may be important in the placement of intakes. We have evidence that the larvae of nearshore sport and commercial fish species are unlike most nearshore larvae and are quite sparse very close to shore where the intakes are. Because of this peculiar distribution, we estimate the loss of sport and commercial fish production, owing to larval mortality via the intakes, to be only 20 tons per year, rather than 160 tons per year as previously expected, thus reducing the predicted impact to one that is relatively minor.

Diffuser losses: We estimate that relatively few fish larvae will be killed by turbulent shear, and believe that this will be a minor effect. We also do not expect the larvae of sport and commercial species to suffer translocation mortality in the plume. However, translocation may cause very large losses of fodder fish larvae.

4. Upwelling Effects of SONGS

SONGS' diffusers will bring extra nutrients to the surface, and move them offshore. This could result each year in the production of roughly 460 tons of anchovy. We believe this will have a negligible effect on sport and commercial fish production, and virtually no effect on nearshore sport and commercial fish production.

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Table 1. Summary of predicted effects of SONGS Units 1, 2 and 3 upon nearshore fish species. Numbers are metric tons per year.

	In biomass	In production	In sport and commercial production
(1) Losses by direct impingement of juvenile and adult fish in intakes			
Fodder fish	31-51	25-41	0-4
Sport and commercial fish	7-12	4-7	4-7
Electric rays	7-13	5-8	---
Other fish	5-8	3-5	---
		Subtotal	4-11
(2) Losses by kill of planktonic stages in intakes			
Fodder fish	358	287	3-29
Sport and commercial fish	34	20	20
		Subtotal	23-49
(3) Damage to kelp bed	0-9	0-3	0-3
		TOTAL	27-63

KELP

Introduction

Kelp beds constitute a distinct and important habitat in the nearshore marine ecosystem in Southern California. Over 760 species of animals (invertebrates and fish) and over 120 species of plants have been found in kelp beds in Southern California. At least two fish species (kelp perch and kelp clingfish) are rarely found outside of kelp beds, and many invertebrate species occur most commonly in this habitat. In the San Onofre kelp bed (SOK) alone we have recorded 164 species of animals and 16 species of plants - certainly an underestimate of the actual diversity. In the three local kelp beds (SOK, San Mateo kelp and Barn kelp) we have recorded 384 species of animals and 36 species of plants. Kelp beds are highly productive of sport fish, including the highly valued kelp bass.

Kelp plants grow very rapidly, and as plants die, or parts of plants break off, they produce food for bottom-dwelling animals. In December 1978, for example, SOK produced an estimated 9 tons of detritus per day.

San Onofre is in an area where kelp beds are (now) rather scarce. However, the local beds maintain ecological continuity between the more extensive beds to the north and south.

Historically, San Onofre kelp bed has exhibited two states: (a) the "normal" state in which much of the available rocky substrate is covered by kelp as is now the case, but the degree of cover varies; (b) periods following catastrophic die-offs of adult plants, during which the bed is non-existent, at very low coverage, or is recovering.

Predictions

(1) It is likely that SONGS Units 2 and 3 will alter the normal state by reducing the density of kelp plants in the offshore portion of the bed. This is the major area of the bed. The reduction could be very small or very large. There are several confounding factors which prevent us from stating a most likely extent of reduction in abundance at present.

(2) SONGS probably will lengthen the periods during which the bed is absent, or very sparse, following catastrophic die-offs.

(3) We expect to see some reduction in the abundance of shrimp species in the canopy in a portion of the kelp bed. No quantitative prediction is possible. This change could alter the diets of fish in the bed.

Mechanisms

Turbidity: SONGS will affect the bed mainly by increasing the turbidity downstream from the points of discharge. This increase will be small in summer, but in spring it is predicted to lower light levels in the water column. The reduction at the bottom in the offshore portion of the bed is predicted to be about 40%. The lower light intensities that result will probably reduce the frequency of successful recruitment of young kelp plants. It is also likely to reduce the growth of kelp plants. Both effects are likely to reduce both the biomass of kelp in the bed and the number of plants.

Fouling: SONGS' plumes are also likely to increase the degree of fouling of kelp plants by various invertebrates that settle on to and live on kelp. Increased turbidity, and perhaps turbulence, are among the mechanisms that could increase fouling. Fouling is likely to 1) decrease the rate

of kelp growth, 2) increase the rate of loss of parts of the plant, and 3) perhaps increase the death rate of plants.

Sea Urchins: Urchin populations may also be increased because SONGS will increase the supply of particulate organic matter that the urchins can use as food. Our studies show that urchins kill a large fraction of kelp plants in parts of the bed, and they probably also interfere with recruitment by grazing on small, young kelp plants.

Sedimentation: The operation of SONGS is not expected to alter the sedimentation rate in SOK.

Temperature: Temperature changes caused by the SONGS plume will be small and are not likely to affect the bed significantly.

Nutrients: Part of the time, the concentration of nutrients may be somewhat increased in the water surrounding adult kelp plants, as a result of upwelling via entrainment. This may increase the growth rate of kelp plants.

Competitors: When kelp is removed from the substrate other plants and animals can grow in its place. These organisms may prevent or slow the re-colonization of kelp, by taking up the space. Although we have information on these organisms, it is not possible to predict whether SONGS will significantly influence these interactions.

Toxic Substances: During the course of the studies at SONGS, circumstantial evidence has been found for the existence of toxic materials in the discharged water from Unit 1. We can make no definitive statement as to whether or not such toxic substances will be discharged by Units 2 and 3, except that chlorine will continue to be used on an intermittent basis.

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MYSIDS

Introduction

Mysids are small shrimp-like creatures that live in shallow water just above the ocean floor, or amongst kelp canopy and other benthic algae. At night some of them rise several meters into the water column, and at this time they are more likely to be entrained by SONGS. Unlike true plankton, they can swim against weak currents, and so can maintain their position to some extent.

Mysids were chosen as a target organism for several reasons.

1) They have similar biology to a number of other groups of "hypo-plankton" that live close to the bottom.

2) They are important food items for a number of fodder fish (e.g. queenfish), which in turn are fed on by sport and commercial fish.

3) Like a number of plankton species, some mysid species live only close to shore and will be taken into the SONGS cooling system and will also be transported offshore by the diffusers. However, since they have a longer generation time than plankton, they are likely to recover more slowly from such extra mortality, and are therefore more likely to show local depressions in density. Mysids are therefore expected to be a good "marker" group for the effects of SONGS.

Predictions

1. Our mysid studies indicate that we should see a reduction in density of about 50% for several kilometers away from SONGS, and smaller depressions on the order of 10 km long. There are several factors that prevent us from

being certain about these effects. First, we are forced to make assumptions about the numbers killed by the diffusers, since we cannot measure this loss. Second, *we do not know how strong compensation will be.*

2. SONGS intakes will kill several billion mysids per year, weighing 50-60 tons. The diffusers could kill several hundred tons of mysids. If, for example, 10% of those entrained by the diffusers were killed by being carried offshore to unfavorable habitat, the annual kill would be rather less than 200 tons. We are unable, at the moment, to give a most probable estimate of diffuser losses.

Mysids constitute about one-half of the total of epibenthic organisms that are subject to entrainment. A similar mortality rate for all of this group would thus give an annual kill of all organisms of this type of about 350 tons.

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If these 350 tons were lost to the fodder fish, we could expect an annual loss of fodder fish production on the order of 30 tons. However, the MRC fish study group believes that much of the mysid biomass killed and moved offshore will be eaten by these same fish species in the region of the diffusers. Some mysid material will, of course, fall uneaten to the ocean floor. There it will join food webs that lead in part to benthic fish. These food webs are less efficient than the mysid → fodder fish chain, so we could expect some overall loss of fodder fish production, although much less than 30 tons per year. We do <u>not</u> predict, therefore, that the mysid losses will have a significant effect on sport and commercial fish production.

PLANKTON

Introduction

The plankton is made up mainly of small drifting organisms that are generally moved about passively by currents. Phytoplankton are single-celled plants that form the basis of most animal production in the oceans. Zooplankton are small animals, some of which can swim actively and control their movements to some degree. They include the meroplankton, such as clam larvae, which are the planktonic stages of bottom-dwelling organisms, and holoplankton, which spend their entire life in the plankton. The predictions focus on the plankton as a balanced indigenous community, and as food for fish.

Predictions

1. The plankton studies have established that some zooplankton species are restricted close to shore (within 3-4 km), and it is probable that SONGS will reduce the local density of this group. It is probable that there will also be changes in the relative abundance of species in the zooplankton assemblage in the inner nearshore zone. The magnitude and extent of these changes cannot be predicted, and will depend on mixing rates, the ability of the populations to compensate, and on interactions between species. As an indication of the likely scale of the effects, we expect them to be somewhat less extensive than the predicted mysid effects.

2. SONGS' intakes probably will kill on the order of 10 trillion of the larger zooplankton per year, weighing about 1200 tons. Most of the zooplankton withdrawn at the intakes will enter the benthic food chain and will be lost as a direct food source for fodder fish. The fate of these diverted zooplankton is discussed in the Soft Bottom Community predictions.

We cannot yet estimate precisely the kill of plankton entrained by the diffuser plumes. If 10% of those entrained were to be killed by being moved offshore to unfavorable habitat, the annual kill would be on the order of 4000 tons. This transported plankton will be eaten largely by the same species of fodder fish that would have eaten it inshore, before SONGS began operation. We therefore do not expect to see significant changes in the overall abundance of fodder fish or sport and commercial fish as a consequence of this shift in biomass.

3. About half of the time, the diffuser discharges will bring to the surface, offshore, relatively nutrient-rich water from closer to the shore and nearer the bottom. We estimate that this will result in the annual production of an extra 84,000 tons of phytoplankton in the mid or outer nearshore waters. The fate of this extra biomass is discussed in the Fish predictions.

SOFT BOTTOM COMMUNITIES

Introduction

The soft benthos community is made up largely of invertebrates (worms, clams, crustacea, etc.) that live in and on the sand, silt and mud bottom. These bottom types cover roughly 80% of the area in the general San Onofre region. The distribution and abundance of these species is strongly influenced by the physical characteristics of the sand, silt and mud and by the amount of food material in the area. The communities close to shore (out to a depth of about 10 meters) are less diverse and less abundant than those further offshore. Most of the species are planktonic in their early stages. Although these communities are not as productive of fish, on a per area basis, as are reefs and kelp beds, because they are so extensive they help to support large populations of fodder fish and hence of sport and commercial fish species.

Predictions

1. SONGS Units 1, 2 and 3 will alter the bottom sediments. Close to the diffusers (within 1 km) the sediments will be coarsened and enriched. Beyond this area, in a pattern and at distances that we cannot yet predict, the sediments will become somewhat finer, and they will be enriched. The general result of these changes will be an increase in the abundance, number of species, and, probably, in annual production of biomass in the enriched region.

2. SONGS could have a negative influence on the soft benthos community by killing some of the organisms that live on the bottom but that occasionally rise into the water column. (This group of organisms bears a broad similarity

to mysids.) It will also reduce the number of larvae of some species available for settlement, by killing the early stages that float in the plankton. This could affect the adult density of some species, especially those living in the intertidal and shallow water zones. Among this group, lobster is a sport and commercial species. However, too little is known about the population dynamics of the early stages to hazard a prediction about possible effect on adult densities. We suspect it will not have a significant effect on the overall production of the community.

3. The enrichment of the soft benthos is not expected to influence the production of sport and commercial fish.

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HARD BOTTOM COMMUNITIES

Boulders and reefs near SONGS are covered by a variety of organisms in addition to kelp. These include smaller species of algae and sedentary animals that permanently attach to the rock surfaces. Apart from their intrinsic value as part of the community, these organisms provide both a source of food for fish and important habitat structure, and they may compete for attachment surfaces with kelp.

There are distinct inshore (intake depth) and offshore (around SOK depth) communities. Turbidity is higher inshore, and inshore species are more tolerant of this higher turbidity. They also grow more rapidly than offshore species. It is thus possible that increases in turbidity in the offshore portion of SOK will lead to a change in the community such that inshore species will tend to replace the resident offshore species. Conceivably these inshore species could also slow the recruitment of kelp by outcompeting it for space.

While these possibilities exist, there is no strong evidence to suggest they will occur.

RATIONALE

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FISH CONTENTS

In this section on fish we do not give a separate rationale for each prediction, since the same types of analyses underlie predictions 1 and 2.

A. The affected fish species

SONGS Units 1, 2 and 3 are most likely to have a significant effect upon fish species that live as adults mainly nearshore (within about 4 km of shore), and that produce planktonic (drifting) eggs and larvae in the same zone. Most species of fish in the SONGS area are of this type. However, most individuals, and most of the total tonnage of fish are Northern anchovies. Anchovies also extend well offshore. There are several hundred billion anchovies in the California Bight, they move enormous distances, and SONGS will not significantly affect the population of this abundant species, although the Plant will kill large numbers of anchovies. They are not considered in most of the analyses below (but see Section I), which concern nearshore species only. A numerically small group of nearshore species either carry their young internally, or have planktonic larvae but lay attached, not free-floating, eggs. This group is also excluded from subsequent analyses.

We will be concerned mainly with those nearshore fish species that produce both planktonic eggs and planktonic larvae. These species fall into one of two groups. (1) Forage or fodder fish. These species eat plankton, small bottom-dwelling organisms, mysid shrimps, etc., and are themselves food for sport and commercial species. The major species in this category are queenfish (Seriphus) and white croaker (Genyonemus).

(2) Sport and commercial fish are the second group. Among nearshore species, halibut and white seabass are the main commercial species while kelp bass and sand bass, and halibut, are the main sport species. These four

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species made up over three-quarters of the 1975 sport and commercial catch of nearshore fish in the fish blocks near SONGS.

B. Mechanisms

There are six known or suspected mechanisms through which SONGS can affect fish populations. These are:

(1) Killing juvenile and adult fish as they are taken into the intakes of the cooling system (via impingement and entrapment).

(2) Killing planktonic eggs and larvae that are taken into the intakes.

(3) Killing planktonic eggs and larvae that are caught up (entrained) by water jetting out of the discharge or diffuser systems.

(4) Loss of fish from special habitats (e.g. kelp).

(5) Loss of fish food that is moved by the cooling system.

(6) (Sub)lethal effects of discharged organochlorines.

We have no evidence that mechanisms (5) and (6) will operate to affect sport and commercial fish production, and they will not be discussed further in this Report.

C. Estimation of probable losses of fish

(1) Direct kill of juveniles and adults in intakes

Unit 1 kills, on average, 16.7 tons of fish per year. The fish are disposed of on land. Of these fish, 10.2 tons are fodder fish, 2.5 tons are electric rays (which are of scientific and economic importance), 2.4 tons are nearshore sport and commercial fish species, and 1.6 tons are other species.

The intake structures of Units 2 and 3 have been modified to reduce the fraction of fish taken in by the intakes. In addition, a fish-return

system has been devised to return those caught back to the ocean. This system has not been tested. The MRC fish study group feels that the fish-return system is likely to kill or fatally injure most fish that pass through it. If the new systems are 50% efficient, the total intake mortality will triple. If they are completely inefficient, total intake mortality will increase about 5-fold since all three structures provide about five times as much attractive "reef structure" as Unit 1. (The volume of water taken in by all three units will be six times that taken in by Unit 1.) If the fish-return system is not more than 50% efficient, the annual impingement fish kill will fall between 3 and 5 times that of Unit 1, or 50-84 tons, of which 7-12 tons will be nearshore sport and commercial fish. This is equivalent to 4-7 tons of nearshore sport and commercial fish production.

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The losses to Unit 1 already produce measurable effects on queenfish. The population of this species within ½ km of the intake (and perhaps as far as 2 km) has fewer young fish and fewer females than more distant populations. Young and female fish are precisely the groups taken in selectively by the intakes. Two-thirds (by weight) of the fodder fish taken in are queenfish. Some 31-51 tons of fodder fish will be impinged. These fish would otherwise have contributed 25-41 tons of fodder fish production (Table 1).

(2) *Killing of planktonic fish eggs and larvae in intakes*

Most nearshore species spend 2-4 months as planktonic eggs and larvae and throughout this stage can be caught up by the intakes or diffuser water. This is the major source of mortality. It is estimated by a somewhat complex procedure involving a model of fish mortality, and we describe the methods only briefly. There are a number of steps in this procedure.

(a) The density of eggs and larvae of various ages, in water at various depths and distances offshore, is estimated from samples. (There is a tendency for older larvae to occur inshore and nearer the bottom, at diffuser and intake depths.) Next, the rate at which SONGS will withdraw water from each of these locations is estimated (from a model of SONGS hydrodynamic behavior). This gives the number of eggs and larvae that will be entrained. Finally, an assumption is made about the fraction of entrained eggs and larvae that will be killed. All of those passing through the intake are assumed to die. (Similar calculations can be made for those caught up by the diffusers, but we cannot yet estimate the fraction of those taken up that will be killed.)

These various estimates allow calculation of the expected number of eggs and larvae that will be killed per unit time (say, each day), immediately after the Plant is turned on (Fish Appendix 1).

We cannot assume this kill rate will continue indefinitely. For example, some water that has been affected by the Plant may remain in or return to the vicinity and mix with "new" water that moves into the area. When this happens, the local density of eggs and larvae will be lower than elsewhere, and fewer eggs and larvae will be killed per unit time.

A detailed model of the current regime in the SONGS area could be used to estimate the rate of replenishment of water in the area, and hence the local density of eggs and larvae exposed to SONGS. Such a model was not available when the present calculations were made.

(b) Instead, a model was used that simply assumed that SONGS will draw eggs and larvae only from some specified region along the coast. Inside this region, all eggs and larvae are assumed to be equally vulnerable (good mixing

is assumed). No egg or larva outside the region can be killed by SONGS and no eggs or larvae can leave the region. The model has the following features (Fish Appendix 1):

• Eggs are produced in this region at a constant annual rate that is the same as elsewhere. (This is essentially the conservative assumption that, even if SONGS kills many plankters and subsequently lowers adult density in the region, reproductive fish will move in from elsewhere.)

• The model calculates the chance that an egg or larva of a given age, within the region, is killed by SONGS before it reaches the next age class (which is 2.5 days older). This is done for all age classes up to the point when the larva becomes a juvenile (4 months in queenfish, for example). Since eggs and larvae die off extremely rapidly due to natural causes, most of them are not killed by SONGS but die of natural causes. This natural death rate is taken into account by the model.

• The chance of any individual being killed by SONGS before it moves out of its age class depends on the size of the region chosen (the chance is smaller when the region is bigger because within 2.5 days a smaller fraction of the water in the region passes through SONGS). Clearly, if a very small region is chosen, a given individual can be exposed to risk on different occasions since the same parcel of water passes through SONGS many times. In this case, the density is rapidly depleted, the fraction killed is high, and most larvae do not grow very old. On the other hand, the number killed is somewhat smaller.

• Since the natural mortality rate is high, there are always far fewer older larvae than younger larvae and eggs. This is reflected in the predicted

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SONGS kill. For example, under one set of assumptions, SONGS will kill in a year 16 billion eggs and 4 billion larvae of nearshore fish.

Clearly the choice of the size of the "affected region" is somewhat arbitrary. Choosing a very small region (say 1 km) is equivalent to assuming virtually no currents along the shore, and hardly any replenishment of the waters around SONGS by "new" water. This will overestimate the degree of local suppression, but will underestimate the number killed - larvae from elsewhere that in reality would get to SONGS are not counted. On the other hand, choosing a very large region (say several hundred kilometers long) is equivalent to assuming that fish eggs and larvae move huge distances in their lifetimes. This would maximize the number killed, but (especially since thorough mixing is assumed) it would spread the effect out very thinly. We feel that this latter scenario is closer to the real situation. 50 km was chosen as a compromise between smaller regions within which complete mixing can be assumed, and larger regions within which all doomed fish larvae are certain to have been produced. SONGS will kill billions of eggs and larvae, and the degree of movement of eggs and larvae will determine whether there is a pronounced local depression or a less obvious, but much more extensive, depression. If there is no re-entrainment of "old" water by SONGS, a choice of 50 km will underestimate the number of eggs and larvae killed.

The result of the model's calculations is a predicted number of eggs and larvae killed per year (breeding season) in each age class.

(c) These predicted losses of eggs and larvae are then converted into an equivalent number of 13 month old fish (Fish Appendix 1). (An age of 13

months is chosen primarily because this corresponds in size to that of the average fodder fish eaten by sport and commercial fish.) The idea involved in calculating 13 month old equivalents is as follows: an egg has roughly 1 chance in a million, under natural conditions, of becoming a 13 month old adult. Therefore, if SONGS kills an egg, this is equivalent to killing only one-millionth of a 13 month old fish, because in all likelihood the egg would have died anyway. However, if SONGS kills a 4 month old larva it has killed the equivalent of .4 of a 13 month old adult, because a 4 month old larva under natural conditions has a 40% chance of becoming a 13 month old adult. It is predicted that SONGS will kill the equivalent of several million 13 month old adults of nearshore fish species.

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At the moment, age distributions of larvae are available for only the two major fodder fish species. To estimate losses of sport and commercial species we have therefore assumed that, averaged over the season, the sport and commercial species have the same age distribution as these two fodder fish species. The estimates of sport and commercial losses owing to larval mortality therefore are based on this, as yet untested, assumption.

(3) <u>Diffuser losses</u>

(a) Turbulent shear losses

There is evidence from the literature that fish larvae die when they are subjected to shear forces on the order of several hundred dynes/cm^2 over a period of several minutes. Losses due to this mechanism were estimated in two steps (Fish Appendix 1). First, the fraction of secondarily entrained water that is likely to be subjected to shear forces on the order of 100/cm^2, or greater, was calculated. Second, the number of larvae subjected to this

stress was estimated from known larval densities and from the estimated amount of water entrained. These calculations suggest that only a relatively small number of larvae will be killed in this way.

(b) Translocation losses

Nearshore fodder fish larvae show a very clear pattern, in which density falls off very rapidly several kilometers from shore. The pattern suggests that larvae that are carried farther offshore die. During some parts of the year, SONGS' diffuser plumes are expected to move some inshore water to an area 5 km or more offshore.

The larvae of sport and commercial fish species extend from close to shore to about 7 km offshore. We therefore do not expect SONGS to cause translocation mortality in this group.

At some times of the year, especially when they are older and "more valuable", the larvae of both queenfish and white croakers do not extend beyond 2 km from the shore. We therefore expect large translocation losses of fodder fish larvae, but we are not able to make a quantitative prediction. Some idea of the possible magnitude of these losses can be gained by noting that if 10% of larvae entrained by the diffuser plumes were to be killed, total fodder fish losses would roughly double.

(4) <u>Losses from damage to kelp bed</u>

Damage to the kelp bed and its biota may be anything from negligible to extreme (see Kelp Predictions).

D. <u>Conversion of losses to biomass (weight of standing stock of fish)</u>

The losses of 13 month old "adult-equivalents" were divided between sport and commercial fish and fodder fish according to the frequencies of these two

types in the larvae affected. Among nearshore planktonic spawning species, in general, four-fifths of the larvae are fodder fish and the remaining one-fifth are sport and commercial fish. However, their relative frequencies vary with proximity to the shore and with position in the water column, and these differences were taken into account.

Next, numbers lost were converted to a weight (biomass) for each group (sport and commercial fish live longer than fodder fish and are larger, so the conversions are different) (Fish Appendix 1). The idea here is that, once SONGS has been operating for several years, 1, 2, 3, . . . year old fish are all affected and each year there will be an average loss of fish weight, spread over all ages, in each species.

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E. Conversion of losses to annual production

Each year, each fish population produces a certain tonnage of "new" biomass, through reproduction and growth. In a perfectly balanced fishery, each year this same amount of tonnage would be consumed - by natural deaths plus the fish harvest. The annual production of a typical sport and commercial population is reckoned to be about 60% of the standing stock (biomass). Thus, when the equivalent of 100 tons of sport and commercial biomass is lost as larvae and eggs, this is equivalent to a loss in production of 60 tons. Similar calculations are possible for fodder fish, where the figure is thought to be 80%.

F. Conversion of fodder fish losses to sport and commercial losses

Sport and commercial fish depend predominantly on fodder fish and, since the biomass of the latter is expected to be reduced, there should be less food for sport and commercial fish. It is difficult to know how to estimate the

effects on sport and commercial species of this predicted loss of fodder fish production. A standard rule of thumb is to assume that 10 pounds of fodder fish production yields one pound of sport and commercial production - a 10% "transfer efficiency". However, if sport and commercial fish population are being held at relatively low densities, say by fishing (Section G), then changes in food supply may have little or no effect on their production. In addition, the fodder fish losses may be partly or largely compensated for (see next section). These considerations suggest that 10% is too high a figure. We think it unlikely that sport and commercial fish production is totally unrelated to fodder fish production, and so assume a 1% relationship as a lower (and more likely) bound.

G. Compensation and declines in nearshore fish species

It is possible that reductions in larval fish density caused by SONGS would lead to higher survival of the remaining fish larvae (for example, by making more food available to each larva). There is, at the moment, no good evidence for such compensation in marine larval fish, and there are a priori reasons for suspecting such compensation would at best be weak. First, fish larvae are already very sparse. Second, it is likely that "chance" (density independent) factors dominate the mortality of these small organisms. Third, much of their food will be killed along with the larvae themselves.

Another possibility is that juvenile or adult fish might survive, grow, or reproduce better in response to lowered density of juveniles. We think this is possible for fodder fish because there is no evidence that their numbers have been declining. However, we think it unlikely that compensation in nearshore sport and commercial fish would be adequate in the face of significant

extra mortality. The main reason for this view is that these species appear to have declined in Southern California since the mid-60s (Fish Appendix 1).

The evidence for declines in nearshore sport and commercial fish species is by no means unequivocal. We have to rely on indirect measures of fish stocks. The major evidence is from California Department of Fish and Game records of sport and commercial catches. These suggest strongly that halibut, in particular, has declined, that kelp bass and sand bass may have declined, and that the more desirable nearshore sport and commercial species as a group have declined.

Several arguments can be made against these conclusions. Counter-evidence, together with comments, is as follows:

(1) Populations fluctuate naturally, and these species showed strong declines in the 1950s, followed by a recovery.

Populations do fluctuate. But this is not evidence that current declines are "natural" and can be ignored. The declines in the 1950s, for example, may have been caused by loss of kelp bed habitat, and DDT in the Bight, and these two mechanisms are now diminished.

(2) Catches of fish in power plants do not show clear evidence of declines.

However, the data from impingement by power plants suffer several defects. First, such data are highly influenced by catchability of fish (which is influenced by annual variations in the weather), as well as by their density. They usually are available for only a few years in the 1970s, and such variations in catchability could easily obscure real trends. Second, the data are extremely variable, and this could obscure trends over this short period.

Third, the data _are_ for only the 1970s, often not for the whole decade, and the Fish and Game data show that the decline was most precipitous in the mid to late 60s and has been rather slight in the 1970s. (The Fish and Game data are _much_ less variable than the Power Plant data, especially in the 1970s.) Thus, we would not even necessarily expect to see a decline in these Power Plant data.

On balance, we believe the data support the conclusion of a decline in desirable nearshore sport and commercial fish.

H. _On-shore off-shore water movements_

The predictions have not taken into account the possibility of large scale onshore and offshore movements of water. (MRC is now measuring this phenomenon.) Such movements could create "circulation cells" that would slow down the longshore movement of eggs and larvae (although it is possible that, by choosing water layers, larvae could escape from such cells). This would reduce the estimated loss of larvae, but would create a more detectable local depression in larval density around SONGS.

I. _Upwelling caused by SONGS_

Some of the water entrained by SONGS' diffusers will come from below 7 m depth. Water at this depth in the region of the diffusers is rich in nutrients, but has low light levels, so that it produces little phytoplankton. The diffuser plume will generally move this (and other inshore water) closer to the surface, where there is more light, and farther offshore. This will result in an absolute increase in phytoplankton production in this region.

We estimate (Fish Appendix 2) that, each year, some 84,000 tons of additional phytoplankton will be produced. Most of this will be eaten by

zooplankton. Although it is not possible to say exactly how this production will pass up the food chain, a reasonable estimate is that half of the phytoplankton will be eaten by microzooplankton, then by macrozooplankton, and then fodder fish. The other half of the phytoplankton will be eaten by macrozooplankton, and then by fodder fish. In this region (roughly 3½ km offshore) the major fodder fish is the anchovy, and most of the new production should pass to this species. A transfer efficiency of 10% would produce, in tons of fodder fish:

$$[4.2 \times 10^4 \times 10^{-2}] + [4.2 \times 10^4 \times 10^{-3}] = 460 \text{ tons.}$$

During these transitions the new production (as phytoplankton and zooplankton) will be moved away from the area of production and thoroughly mixed. The anchovy population is also extremely mobile and well mixed, so this production of anchovies would be expected to be spread over a very large fraction of the Bight population.

460 tons is a miniscule fraction of yearly California anchovy production, which is about 1-2 million tons. We believe it would not result in any real increase in yield to sport and commercial fish. It should be remembered that we have made a similar argument for ignoring anchovy losses: each year, SONGS will kill on the order of 10 times as many anchovy larvae as other fodder fish larvae, and the fodder fish losses themselves are equivalent to more than 300 tons of production, but we predict no effect from these losses. Clearly, in "production equivalents", the anchovy losses are much greater than 300 tons, but we believe it is sensible to assume that perturbations of this order, spread over the whole anchovy population, will have no effect on adult anchovy standing stock, and hence production.

As discussed in Section F, the transfer efficiency from fodder fish to sport and commercial fish probably lies somewhere between 1% and 10%, and we have argued it is likely to be close to 1%. If the increase in anchovy production were to be passed on, we would expect it to produce an extra 5-46 tons of sport and commercial fish, and believe the lower figure much more likely. Most of this production would not be in nearshore sport and commercial fish, since the mass of the anchovy population is offshore.

KELP CONTENTS

I. Biology of Kelp

We begin by looking at the basic population dynamics of the San Onofre kelp bed.

(A) "Normal" conditions

It appears that, even in the absence of catastrophic events, the kelp bed is rarely in a "steady-state" or equilibrium condition. It is instead dominated by physical and oceanographic conditions that are highly variable. In the present study (1976 to 1980), only by the end of 1979 did SOK cover most of the cobble substrate available. Naturally, the amount of kelp (number of plants and areal extent) on any section of the bed fluctuates in response to changes in bottom conditions, storms that tear adult plants from their sites of attachment, water temperature, availability of light and nutrients, grazing by sea urchins and probably fish, fouling, and periodic recruitment. Patches of kelp within the bed increase and decrease and even disappear and reappear under normal conditions.

Recruitment of new plants is a major dynamic event that is episodic, in response to seasonal and annual variation in physical and chemical conditions. It appears that recruitment occurs, on average, only once every three years. (However, recruitment rate has been examined, in this and other studies, for a total of only 12 years or so.) Although kelp has a complex life cycle (Figure 1), for present purposes there are only two important processes affecting recruitment of adults: (i) the ability of the tiny male and female stages (gametophytes) to reproduce and hence produce the microscopic first stage of the actual kelp plant (sporophyte); (ii) the ability of juvenile plants to grow up into adult kelp plants. Experiments have shown that light

is an essential factor (but not the only factor) controlling these two processes.

We need to look briefly at the dynamics of the life cycle.

(1) Reproduction, and recruitment of juvenile plants

The adult plants produce minute propagules (zoospores) that settle on the bottom and become either tiny male or female stages called gametophytes. Each adult plant produces extremely large numbers of these propagules, perhaps continually throughout the year. Thus it is probable that there are gametophytes present, most of the time, in abundance, on suitable areas of the bottom close to adult plants. The critical factor is the occurrence of a combination of suitable physical conditions (including, at least, adequate light and nutrients) that allow gametophytes to reproduce. The gametophytes that do reproduce, produce microscopically small kelp plants. This type of life cycle is known as alternation of generations. In kelp the microscopic gametophytes are the sexual stage. The sporophyte (the actual kelp plant) is the asexual stage. It is also microscopically small to begin with, but passes through juvenile and subadult stages to become the massive adult kelp plant.

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Gametophytes are killed by a variety of factors - abrasion, burial by sediments, and grazing by animals - and only a small fraction of them survive to produce sporophytes (Kelp Appendix 1, p. 150). Even so, after a successful reproductive "set", there are thousands of tiny sporophytes per square meter of cobble substrate. Unfortunately, it is extremely difficult to study these microscopically small plants in natural conditions. Quantitative studies have been done only on larger plants that have reached a height of

more than 10 cm (4 inches). At about 40 cm (16 inches) the plant becomes a juvenile (Figure 1). Once again, a variety of factors kill most of the sporophytes before they become juvenile plants.

It appears that the physical environment affects these processes in the following way. Reproduction by gametophytes requires adequate light and, probably, a high concentration of nutrients in the bottom water. When these conditions prevail, the gametophytes absorb sunlight and nutrients each day, until they mature to a reproductive condition. Field experiments show that very few sporophytes ever appear from gametophytes planted out more than 40 days. Thus, in the field, 40 days apparently is the maximum period during which this stage can accumulate the sunlight needed for survival and reproduction. Over this period they need an average of at least .43 Einsteins per m^2 per day (Kelp Appendix 2, p. 5). (Under good field conditions it is likely that the average successful gametophyte manages to accumulate enough light in about 20 days.) The critical question for sporophyte recruitment, in any given year, is therefore: during the period in which gametophytes are present, what is the probability (a) that enough light can be accumulated during at least one 40-day period (called a "light window"), and (b) that nutrients are also adequate during the light window?

It appears that these two conditions co-occur only rarely. (a) The frequency of light windows varies with the situation. In a very sparse part of the kelp bed, where adults were absent and vegetation had been cleared, all of the spring season consisted of light windows (Kelp Appendix 3, Table 1, p. 5). However, in darker portions of the bed, where adults are present in abundance, none of the 40-day periods appeared to have received adequate light

on the bottom. With a light understory of other algae, and heavy adult canopy, about 30% of 40-day periods were light windows on the bottom.

(b) It appears likely that nutrients are adequate only during periods of upwelling. In any given spring these periods last for only a few days, and occur not more than a few times per season (Kelp Appendix 1, Figure E1, p. 260).

Suitable conditions for reproduction occur mainly in the spring, although occasionally also in the fall. It appears that adequate conditions for reproduction occur, on average, only once every three years (Kelp Appendix 2). At any one time the bed is thus generally dominated by a "cohort" of adult plants from a single episode of reproduction.

As discussed below, SONGS is predicted to decrease the frequency at which conditions become suitable for reproduction. We cannot predict whether or not SONGS will affect the number of sporophytes or juvenile plants that arise from any given successful reproductive set. It is likely, however, that some factors will not have much effect on the number produced:

(a) Each adult plant produces enormous numbers of gametophytes. Thus, unless the density of adult plants is catastrophically reduced, we assume that there will be enough gametophytes present to replenish the bed even when adult density is low. (This is equivalent to assuming there is density "compensation" in the survival of these small stages.) There must be some very low density of adult plants at which replenishment through a single reproductive set is not possible, but we make the conservative assumption that it is very low, lower than is encountered during "normal" conditions.

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(b) With respect to light levels, reproduction is all-or-nothing. When adequate light is available, the number of tiny new plants (sporophytes) produced is independent of the light level. The number produced appears, instead, to be associated with the amount of nitrogen in the bottom waters, and this is not expected to be affected by SONGS.

The survival of sporophytes to the juvenile stage is determined by a range of factors (abrasion, sedimentation, grazing).

(2) Survival from juvenile to adult stage

Juveniles frequently suffer a higher death rate than adults (Kelp Appendix 1, pp. 93 and 95), so anything that prolongs the juvenile stage will reduce both the eventual number of adults and the average density of kelp plants. Light affects the growth rate, and so does fouling. These factors are discussed later.

The growth rate of juvenile kelp plants is highly variable. Some plants in a group develop from juvenile to adult in less than three months, while others take more than 13 months. The survivorship from juvenile to adult stage is also highly variable, and depends on, among other factors, both the initial number of juveniles and the number of adults present. The fraction surviving tends to be higher when (a) fewer juveniles are present initially (Kelp Appendix 1, p. 82), and (b) fewer adults are present (Kelp Appendix 1, p. 84, and Kelp Appendix 2, p. 10). These relationships reflect an important result: except when very low densities of juveniles are present, the final number of adults present is roughly constant. (This means there is strong "compensation" or "density-dependence". If some factor reduces juvenile density, the number of adults produced may be relatively unaffected.)

(3) Summary of "normal" kelp population dynamics

A final piece of information completes the picture of "normal" kelp bed population dynamics, namely that the average adult plant survives for about 12 months (Kelp Appendix 2, p. 11). That is, if we start out at some point in time with a cohort of adults produced by a successful reproductive "set" a year or more earlier, we can expect roughly half to die within 12 months. By the end of two years roughly 25% of these adults will remain alive, and by the end of three years, roughly 12½% will remain alive. At this time, on average, we could expect another cohort of adults to appear. In reality, of course, the dynamics would not follow this average pattern, but would vary around it. For example, deaths occur mainly in winter storms, which vary in their severity from year to year; again, reproductive sets will sometimes be spaced one or two years apart, and sometimes four or five years apart.

The number of kelp plants in the bed thus fluctuates, rising rapidly after a successful recruitment event, and declining thereafter. However, the canopy area of the bed will not clearly follow this pattern since the surviving plants will continue to grow. The canopy area can thus increase even though the number of plants may be decreasing.

(B) Catastrophes

We know little about the frequency of catastrophes in the SONGS area before the 1950s. Certainly the kelp beds in the general area were more extensive and continuous when they were observed at various times earlier in the century than they have been since (Kelp Appendix 1, p. 12). It is likely that much of the cobble in this area has been covered by sediments since then. We do not know, however, if the beds were severely reduced between the infrequent

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observations made before 1950.

Two catastrophic die-offs have occurred since 1956 (Kelp Appendix 1, p. 12). The first, in 1958-59, was associated with high summer temperatures (but may have been caused by associated low levels of nutrients). At this time 90% of Southern California kelp beds were destroyed. SOK was not re-established for a period of 12 years (by 1972). In 1976, again a year of unusually high temperatures, SOK suffered a partial die-off, being reduced to less than 10% of its former extent, and only in the offshore segment did plants remain. Recruitment occurred about a year later, and recovery of the canopy took almost two more years.

There are two means by which kelp disperses and, hence, beds recover or become re-established. First, the adult plant casts its microscopic offspring varying distances. Many offspring probably fall very close (a few meters) to the plant. (Observations at SOK show that some offspring may be dispersed one or two hundred meters from the bed, but we do not know if these were offspring from plants attached in the bed, or from plants that became detached and drifted from the bed.) Secondly, adult plants, torn loose in storms, drift and sometimes cast spores on suitable substrate far from their point of origin. Re-establishment of a bed therefore depends on chance events, and seems more likely when a source of "colonists" is close by. This is one reason why the longshore continuity of beds is important. Recovery of a kelp bed that has been drastically reduced, but not exterminated, depends mainly on local reproduction. Observations at SOK, in the very successful reproductive season of 1978, suggest that a large "set" of new plants can arise from quite a sparse kelp bed, and that recovery can be rapid if the catastrophic

die-off is followed quickly by successful recruitment. By contrast, the 1958 catastrophe suggests that major catastrophes can be followed by very long recovery periods because no or extremely few plants survive locally.

II. Estimating the Effects of SONGS Units 2 and 3

(A) Predicted effects on kelp reproduction

The two major factors affecting reproduction are light and nutrients. Increased turbidity caused by SONGS' discharge will reduce the light in SOK during spring, the main reproductive season. The probable effects on reproduction were estimated by first calculating the expected reduction in light and, second, by calculating how this should affect reproduction. SONGS is not expected to alter nutrients on the bottom, where reproduction occurs.

The probable levels of light that will prevail in the kelp bed once Units 2 and 3 are operating were calculated in four steps (Kelp Appendix 1, pp. 222-241, and Turbidity Appendix). First, ambient light levels near the bottom were recorded. Second, a computer simulation model of water movements near SONGS, including those caused by SONGS' intake and diffuser systems, was developed. This was based on information obtained from current meters placed in the ocean near SONGS, and from a physical model of SONGS-induced water movements produced for Southern California Edison. Third, measurements of natural turbidity levels were made in spring and summer. This information allowed prediction of expected levels of turbidity in the kelp bed for these two seasons. Finally, measurements of light and turbidity levels in the field yielded a strong quantitative relationship between light and turbidity. The calculations predict (conservatively) that in spring, in the (most important) offshore half of the bed, subsurface light levels on average will be reduced

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by from 25% to 55%, with a roughly 40% reduction being most likely. No significant reduction in light is expected in the already turbid inshore segment. The offshore half of the bed has been the most persistent during catastrophe, has the densest canopy cover, and constitutes 70% of the total SOK canopy cover. Subsurface light will be much less affected in late summer.

A 40% reduction in subsurface light will reduce the number of 40-day light windows, and hence the probability of recruitment. The amount of reduction depends on the prevailing light regime. In a clear part of the bed, where all 40-day periods are suitable, a 40% reduction in light would cut the number of light windows by 20-30%. At other parts of the bed, where light windows are already scarce, the reduction could be close to 100%. We will use a 20% reduction as a conservative estimate, since the most critical recruitment events occur when the bed is sparse and therefore ambient light levels will be high.

To estimate the potential effect of this reduction in underwater illumination on reproduction, a model of reproduction is useful. A crude model, assuming that <u>only one</u> coincidence of adequate light and nutrients is needed to provide successful recruitment in a season, is as follows. In a season of D days, there is, each day, probability w that the day is the first of a light window, probability n that nutrients are adequate, and probability g that there is an adequate supply of gametophytes. The probability that a given day will initiate successful recruitment is then wgn. If 40-day periods can be treated independently, then the probability that at least one day in the season will initiate recruitment is $1-(1-wgn)^D$ (Kelp Appendix 4).

This model can be used to estimate how a reduction in the number of light windows will affect recruitment. Suppose we reduce the number of light windows to a fraction (p) of their original number (in the case of a 20% reduction, p = .8). The probability a given day will begin a light window then becomes pw, and our model is $1-(1-pwgn)^D$. We assume that only when SOK is destroyed is g < 1, so except when the bed is absent, the model becomes $1-(1-pwn)^D$.

If wgn, or wn, is small, $(1-pwgn)^D \approx 1-Dpwgn$, and the reduction in the probability of successful recruitment will be by a factor close to p. Otherwise the reduction will be less than p. There are three cases: normal SOK population dynamics, SOK absent (when it is destroyed), and SOK reduced (when it is at very low densities).

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In normal times there is very little light in the bed and w is small. Furthermore, those partially shaded areas that do provide some windows suffer a greater than 20% reduction in windows. Thus a 20% reduction seems to be a conservative estimate. Note, with p = .8, the average time between recruitment events increases by a factor of 1/p = 1.25. That is, the average time between recruitment events would be expected to increase from about three years to almost four years.

In the absent phase, g is very small, since recruitment depends on the rare event of a drifting kelp plant dropping spores on suitable substrate. Thus a 25% increase in the time to recruitment is a reasonable estimate. Even in the reduced phase, when w is intermediate and g ≈ 1, n is likely to be very small and the time between recruitment events should increase by 25%.

Overall, therefore, it is reasonable to predict a 20% reduction in the probability of successful recruitment, and therefore a 25% increase in the average time between recruitment events.

(B) Predicted effects on kelp growth and survival

Light and fouling of kelp plants are the major factors that are expected to affect kelp growth. We discussed expected changes in light, above. Here we first describe fouling and then discuss the relationships among light, fouling, and growth and survival of kelp.

Fouling: Several species of small invertebrates settle and attach to kelp plants. Some build tubes from particles in the water, others merely live on the kelp blades. Under normal conditions in SOK, fouling of juvenile kelp plants is rather light, although the fouling organisms are present.

Several experimental studies show that the abundance of these fouling organisms on kelp plants and other surfaces is greater the closer they are to the discharge plume of Unit 1. This increase is caused by (probably several) factors associated with the plume, including increased particles in the water, and increased turbulence which stimulates the planktonic stages of some organisms to settle. It is also associated with lower light levels, but is probably not caused directly by reduced light.

There is evidence (Kelp Appendices 2 and 5) that increased fouling can reduce the growth of kelp plants, and damages them by causing them to lose blades, causing fronds to sink, and attracting fish and other predators.

The relationships between light, fouling and growth were examined in an experiment in which juvenile kelp plants were transplanted to the Unit 1 plume and to other areas in which underwater light levels varied (Kelp Appen-

dix 1, pp. 101-121; Kelp Appendix 2, Table 11; Kelp Appendix 3, p. 6). A multiple regression of growth rate (Δ log length in cm/day), versus irradiance ($E/m^2/d$) and percent cover by Membranipora (a bryozoan that is a major fouling orgsnism), explained 99% of the variance in growth in the experimental juvenile plants at four locations at different distances from the SONGS Unit 1 discharge.

This experiment suggests very strongly that decreases in light and increases in fouling will have a detrimental effect on kelp growth. Unfortunately, the relationships among the three factors (light, fouling and growth) are complex, and this complexity prevents us from making a confident quantitative prediction. The uncertainty arises because (1) the effects of light and fouling on growth are confounded, (2) the relationship between growth and light is different inshore and at SOK, (3) growth and light do not always show a consistent relationship, and (4) we cannot predict quantitatively how fouling will change at SOK.

(1) Lower light was always associated with greater fouling in this experiment, and so we cannot tell how much of the reduction in growth was caused by each of these factors. Fouling alone explained 95.3% of the variance in growth, and light explained 99.5% of the remaining variance, a significant fraction, so we know light has some effect. Light alone explains 99.7% of the variance in growth, and fouling explains 93.1% of the remaining variance (which is not a statistically significant fraction). We have, so far, been unable to separate the effects of these two factors upon growth.

(2) The relation between kelp growth and light in SOK is different from the experimental relationship established inshore. At a given light level

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kelp grows faster in SOK than it does inshore.

(3) There is one pair of observations in SOK that shows kelp growing at similar rates at different light levels (Annual Report, p. 110, Table 4.2).

(4) Fouling appears to be increased by an increasing concentration of particles in the water, and by turbulence. We do not know the quantitative relationships involved, and we do not have a precise prediction for these two variables under SONGS' operation. Furthermore, the organisms may 1) behave differently, 2) be a different mix of species, and 3) differ in abundance at SOK and inshore. Thus, we cannot predict the extent of fouling at SOK once SONGS Units 2 and 3 begin operation.

Experiments now underway should help resolve the relationship between light and growth.

In spite of difficulties of interpretation, however, the transplant experiment predicts that kelp growth will be reduced when SONGS 2 and 3 are operating. Reduced growth would be expected to (a) reduce the average size of plants, and so reduce kelp biomass and cover, and (b) reduce the number of kelp plants. We next explore question (b).

Reduced growth should reduce plant density because death rates of juvenile and sub-adult stages are generally higher than those for adults, and plants would spend longer in the high death rate phases. According to one set of calculations, this would lead to a 70% reduction in the number of plants produced from a cohort of new juveniles (Kelp Appendix 2, pp. 13-17). If compensation operation, the reduction could be as small as 25%.

We cannot place much reliance on these particular figures because different sets of plausible assumptions and relationships give us different

estimates that range from a negligible effect to an even greater than 70% reduction in abundance (Kelp Appendix 5). Furthermore, we still have the problems of the confounding effects of fouling, and one pair of observations of similar growth at different light levels in SOK.

No firm quantitative prediction can be made about growth and survivorship.

(C) Other factors associated with SONGS

(1) Sedimentation

Sedimentation appears to reduce the recruitment of new plants by smothering them and increasing abrasion. However, SONGS is expected to have no effect on the sedimentation rate on the bottom at SOK.

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(2) Sea urchins

Sea urchins (Lytechinus) have caused a large amount (about 45%) of adult mortality in parts of the bed. They also appear to interfere with recruitment by grazing on the microscopic and very small stages of kelp.

SONGS will probably increase the amount of particulate organic matter (POC) at SOK. Schroeter et al. (Kelp Appendix 5) show that urchins grow more inshore than offshore, and argue that this was caused by higher POC levels there. They conjecture that SONGS will therefore increase urchin populations, and hence grazing pressure, in SOK. This seems a reasonable prediction, but we cannot be certain it will occur because other factors (predation, etc.) also affect the abundance of sea urchins.

(3) Toxins

Reduced growth and settlement of various organisms in the Unit 1 discharge plume have led investigators to postulate that the plume contains

small quantities of toxin(s) - perhaps copper or chlorine. Southern California Edison claims that Unit 1 releases extremely small amounts of copper, that copper will be virtually absent from the plumes of Units 2 and 3, and that these units will also use little chlorine.

There are no usable data on toxins from SONGS, and we cannot evaluate their possible role. This point requires investigation.

(4) Temperature

SONGS is expected to have very little effect on water temperatures in SOK (a less than 0.5°C average increase, a maximum of a 1°C increase, and a non-detectable increase over most of the bed).

(5) Nutrients

The concentration of nutrients is expected to increase in SOK in surface and mid waters at some periods of the year. We have no quantitative prediction of this effect, nor do we know the relationship between nutrient levels and adult plant growth. This mechanism could lead to greater plant growth (Plankton Appendix 2).

(D) Overall effects on the kelp bed

The predicted reduction of recruitment, and an increase in mortality, would lead to a reduced density of kelp plants in the offshore portion of the bed. These two effects plus reduced growth of individual plants and greater grazing by urchins would reduce the amount (biomass and cover) of kelp in the bed. Increased midwater nutrients could cause an increase in kelp growth. We cannot make a quantitative estimate of the overall effects.

(E) Effects on shrimp in the kelp canopy

Experiments carried out at various distances from Unit 1 discharge

showed that shrimp densities on settling plates were lower close to the discharge. These spatial differences tended to disappear when SONGS was not operating. It was also shown that the death rate of shrimp in experimental containers was greater closer to the Plant.

The mechanism causing these effects is not known, so no quantitative predictions of the effects of Units 2 and 3 can be made.

Shrimp are important in the diets of various fish species that live in SOK (Kelp Appendix 5, pp. 12-13).

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MYSIDS

1. Annual loss of mysids

From the field sampling program we know how mysid densities change as one goes offshore. Several species, constituting most of the mysid population, are restricted to within 3 or 4 km of the shore (Mysid Appendix 1). Maximum mysid density occurs in the intake zone.

These data, plus information on the rate of SONGS' intake of water, allow us to calculate how many mysids will be taken into SONGS' intakes. Sampling at Unit 1, and laboratory studies, suggest that all mysids taken into the cooling system will be killed.

We are much less certain of the number that will be killed by the discharge plume, which will entrain about 10 times its own volume of water. There are two possible sources of mortality. First, some mysids will die from turbulent shear forces created by the discharging water. We believe this will be a relatively minor source of mortality. Second, some mysids will be carried further offshore in the plume and deposited offshore of their normal habitat. There is as yet no reliable method for predicting the number of mysids dying in this way.

2. Mysid depression

(a) Depression caused by intake and diffuser mortality

If mysid mortality is of the order calculated in Section 1, we would expect there to be a lowering of mysid density downstream from the Plant. The extent and depth of the depression depends upon the rate of mixing with water that has not passed through the Plant, and on the ability of surviving mysids to compensate with increased reproduction, growth or survival.

The probable extent of the depression was estimated using a model that combines a description of water movements and the biology of the mysids (Mysid Appendix 4). The model describes both the ambient current regime and SONGS' plume, and moves mysids about accordingly. It incorporates the natural mortality rate of mysids (as determined from samples) and imposes on this rate the expected SONGS-induced mortality. The model incorporates 100% intake mortality and 20% mortality in the plume. (The model assumed that this was caused by turbulent shear. It is more likely that any diffuser losses will be caused by translocation; however, we use the output as an indication of the scale of possible effects.)

The model predicts that, for much of the year, depressions on the order of 50% should exist out to 5 km or more from the Plant, and that lesser depressions should extend for more than 10 km.

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We need to view these predictions with caution. The model is not a precise description of reality; in particular, it becomes less accurate as it predicts events more distant from the Plant. Also, the amount of translocation mortality is not known. What the model does tell us is that we can expect to see a measurable depression in mysid density, at least several km long, for much of the year, and it probably indicates the maximum size of the depression that could be caused by these mechanisms.

(b) Depression caused by an unknown factor

The Mysid Study group has data suggesting that Unit 1 presently causes a depression in mysid density of almost 50% that extends 6 km downstream (Mysid Appendix 3). This is the difference observed in the longshore pattern of abundance between samples taken when the Plant is on, and when it is off.

There is statistical support for this claim, but there is a difficulty in that the "Plant on" samples were taken in October, while the "Plant off" samples were taken in spring, and the general level of mysid abundance was greatly different at these two seasons. Samples are being taken now, while the Plant is off, to resolve this issue.

The Committee feels there is a further problem with these results. Even if it can be shown statistically that a depression occurs when the Plant is on, but not when it is off, we know of no mechanism that is likely to produce such an effect. (The actual kill via intake and plume mortality would not depress the population for such a distance, and the plume from Unit 1 rarely extends more than 3 km from the Plant.) One suggested mechanism is that organochlorine compounds from the Plant adhere to very small particles and settle out over a distance of 6 km. We have no evidence concerning this mechanism. If the new studies confirm the existence of this depression, further work will be required on this question.

If indeed there is a depression to 6 km caused by Unit 1, then it may be reasonable to expect that the enormous additional kill rate of Units 2 and 3 will extend the depression to 10 km or so. Notice, however, that there is no evidence that the plume from Units 2 and 3 will extend further downcoast than that from Unit 1. Thus there is no certainty that the additional intake and plume losses from Units 2 and 3 would extend an already existing depression.

3. Significance of mysid losses

Mysid populations are extensive along the coast, and our predictions do not imply that SONGS would have a significant effect on the coastal populations. As stated in the "Predictions", we do not expect these effects to have a major

impact on the local populations of fodder fish, although this is certainly a prediction that we need to check when Units 2 and 3 begin operating.

PLANKTON

1. The evidence that some zooplanktonic species are restricted close to shore can be found in Plankton Appendix 1. The centers of abundance of the inner nearshore species are in the areas of the intake and diffusers, and these species are therefore subject to greater SONGS pressures than other less restricted species. Some of the zooplankton restricted to the inner nearshore tend to live closer to the bottom where the longshore currents are slower (MRC Interim Report 1979-02 (II), p. 17), and as a consequence their longshore replacement (mixing) rates could be lower than those for other species. In addition, some of the non-restricted species could be replaced by individuals from farther offshore. All of this would favor the non-restricted species in the recovery from SONGS losses and would tend to promote a shift in relative abundance.

2. Synoptic samples taken in the intake and discharge ports of SONGS Unit 1 demonstrate that few of the withdrawn zooplankton occur in the discharged waters (MRC Interim Report 1979, and Plankton Appendix 2). Presumably, they are consumed during their journey through the intake conduit by the benthic organisms that live on the inner walls. These benthic organisms are purged from the cooling system during heat treatment and reverse flow, and become part of the inshore benthic food chain. The estimates of plankton densities used in predicting intake losses can be found in Plankton Appendix 2.

The estimate of zooplankton entrainment by diffusers is based on total macrozooplankton (zooplankton greater than .2 mm in width) plus the micro-zooplanktonic species _Euterpina acutifrons_. _Euterpina_ was included because

it forms a major part of the diets of most fish larvae and some fodder fish in the area. Using the field samples from 20 dates for macrozooplankton and from 5 dates for Euterpina, the mean concentrations were calculated for different positions expected to be affected by diffuser entrainment (Plankton Appendix 2). This estimate of about 4×10^4 metric tons of zooplankton entrained per year was based on the assumption that equal entrainment occurred at all depths over the full length of the diffusers. The assumption that 10% of those entrained are killed results in an estimate of 4000 metric tons.

Most of the zooplankton biomass moved offshore by diffuser entrainment is likely to be eaten by adult and juvenile anchovies, top smelt, and blacksmiths. According to the MRC Fish Group, the blacksmith should increase in abundance because the diffuser rip-rap provides new habitat and the diffuser plume provides a continual source of zooplankton. In the absence of SONGS these secondarily entrained zooplankton would have been available to the same predators and to the late larval stages of the fodder fish Genyonemus and Seriphus.

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3. The diffuser discharges will result in replacement of part of the offshore surface water by a plume consisting of a mixture of nutrient-rich waters from closer to shore and nearer the bottom. The detailed methods used in estimating the amount of nitrate plus nitrite added to the surface waters, and the conversion of these estimates to estimated phytoplankton production, are explained in Plankton Appendix 3.

SONGS will induce a real net increase over present primary production off San Onofre. First, in surface and mid waters where chlorophyll is high, nutrients are low, suggesting that when nutrients get into the high chlorophyll

waters they are taken up rapidly. Conversely, the presence of deeper waters high in nutrients and low in chlorophyll presumably indicates that the phytoplankton there are utilizing nutrients at a lower rate (Plankton Appendix 3). Therefore the nutrients in the bottom waters upwelled by the diffusers will be utilized at a far higher rate when they reach the surface.

Second, the waters replacing the entrained waters will also be high in nutrients and low in productivity. During periods of moderate to strong longshore currents, entrained water will be replaced primarily from longshore and similar depths. Under very sluggish conditions most of the entrained waters will come from offshore. In both cases, the water will be rich in nutrients (Plankton Appendix 3).

SOFT BOTTOM COMMUNITIES

The basis for the predictions can be found in Soft Bottom Communities Appendices 1 and 2.

1. Probable sediment effects were estimated by establishing the existing statistical relationships among abundance, diversity and characteristics of the sediments. Probable changes in the sediments were estimated (very approximately) from information about the weights of various materials in the SONGS' plumes, from information about water movements, and from information about the settling rates of various classes of materials.

2. Some 17% of the benthic species at some time rise into the water column and are at risk to entrainment by the intake or the discharging water. Too little is known about this group to make a firm quantitative estimate of losses, but we expect them to be roughly the same as mysid losses.

We are very uncertain about possible losses of planktonic larvae and the potential effects. This group of plankton is very poorly known. We do have data showing that the larvae of some intertidal and nearshore species are restricted inshore. However, we cannot estimate losses of benthic larvae because we do not know how to estimate mortality caused by the plume. Finally, although we know for some rocky bottom species that have been studied, that larval settlement far exceeds the number needed to maintain the adult population, we do not know if this is always the case, or if it is true for soft bottoms. If it were, likely reductions in larval settlement would have no effect on adult numbers.

It is possible that some intertidal and shallow water species will show reduced adult densities close to SONGS. However, it seems likely that total

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densities will not be significantly reduced, and that any reduction of a particular species will be made up by increased density of others.

3. The enrichment of bottom sediments should have virtually no effect on the production of sport and commercial fish. The enrichment derives from SONGS' killing of organisms in the water column, and so represents a shift of material. The food chains on the soft bottom eventually lead to the same group of sport and commercial fish species as do planktonic food chains; however, there should be some additional losses of this material as it passes up the benthic food chain.

HARD BOTTOM COMMUNITIES

The Hard Benthos Project (Hard Benthos Appendix) has shown clear differences between nearshore and offshore communities on the underside of experimental panels, and some degree of similarity between the communities on panels and on natural boulders. There is also a correlation between these differences and turbidity; and the inshore species grow faster than offshore species at high turbidity.

We believe there is no strong evidence that major changes will occur in this community. Several factors prevent us from making quantitative predictions, including the lack of close similarity between experimental panels and the tops of boulders, and the lack of quantitative relationships between possible changes and turbidity levels.

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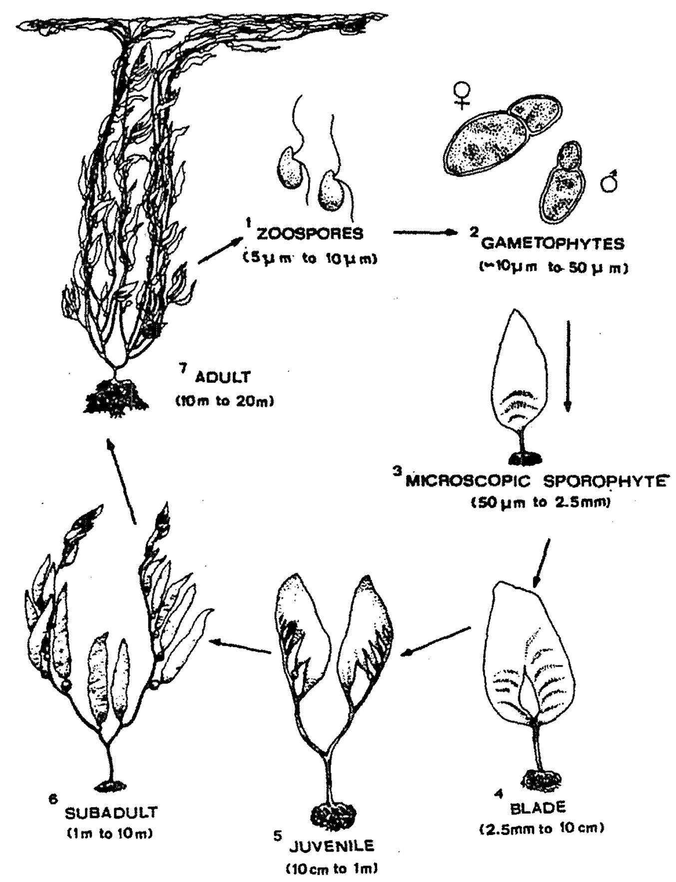
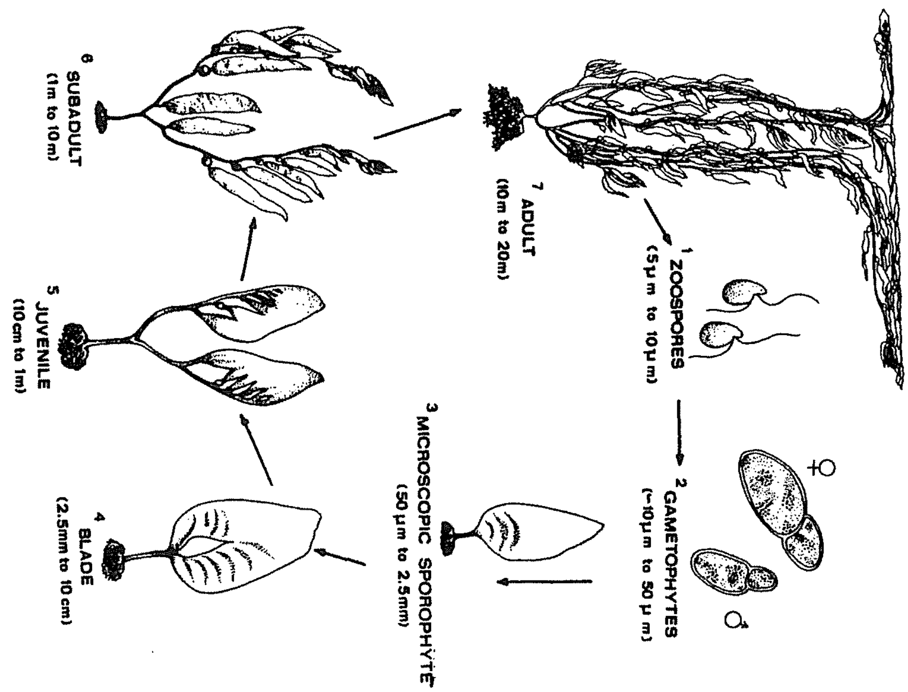




A diagrammatic representation of the Macrocystis life cycle. Stage names and approximate sizes are given.

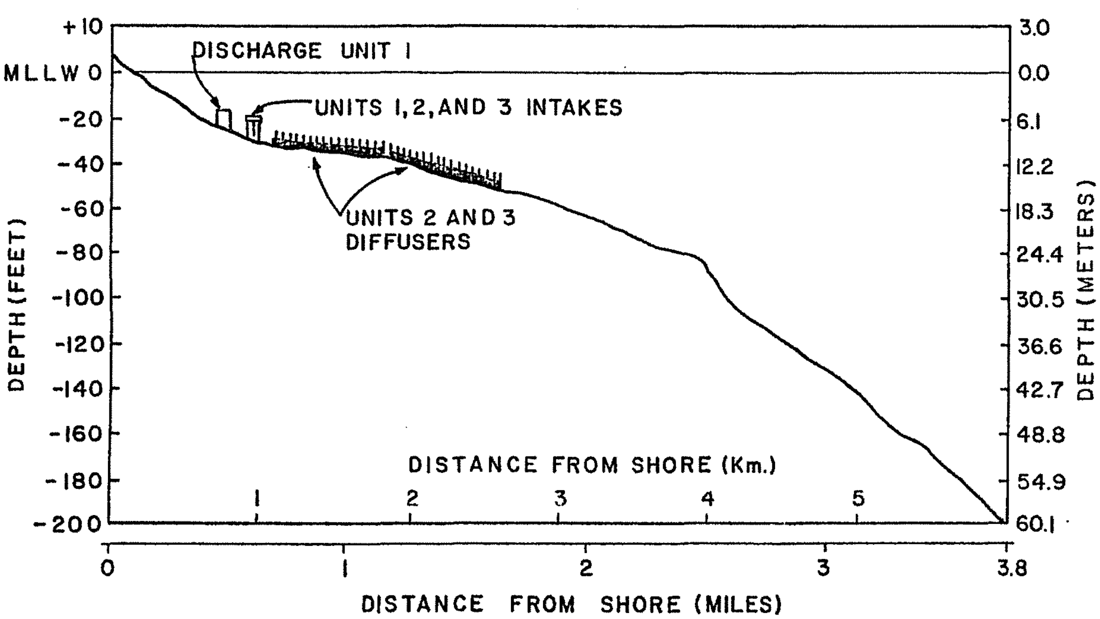
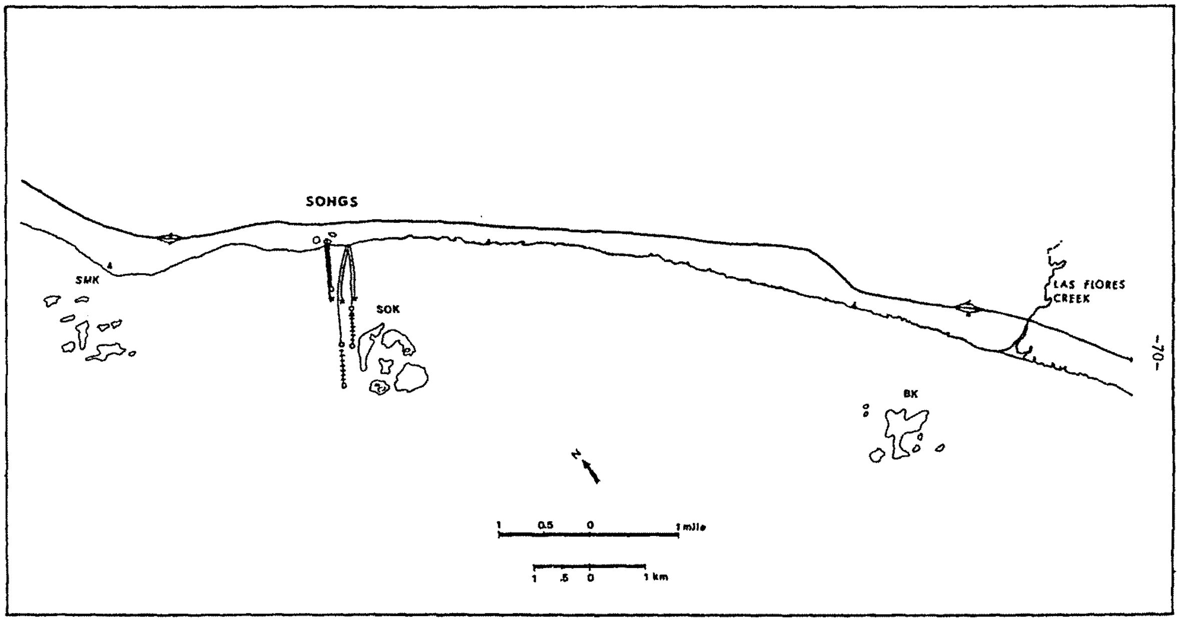
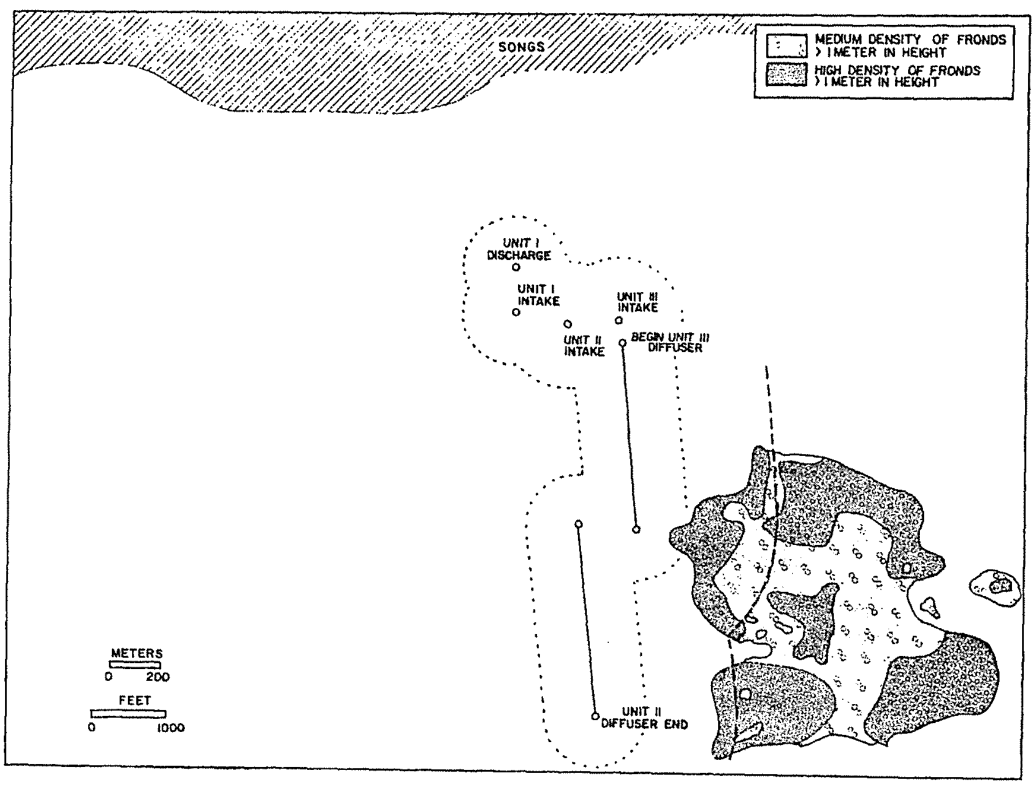


Figure 1. Offshore profile of the cooling system of SONGS Units 1, 2 and 3. (Modified from Figure II-3, Southern California Edison, San Diego Gas and Electric Company, Thermal Effect Study, San Onofre Nuclear Generating Station Units 2 and 3, Vol. 2, September, 1973.)



Map 1. Map of the region near SONGS. The kelp beds shown are the high density portions of San Mateo (SMK), San Onofre (SOK), and Barn (BK) kelp beds, as measured in December, 1978.



Map 2. Map of the coast near San Onofre showing the cooling systems of SONGS Units 1, 2 and 3 and the medium to high density areas of kelp measured in December, 1978, within the San Onofre Kelp (SOK) bed. The boundaries of the areas where the sediments are modified are indicated by dashed lines. The dotted line delimits the area within 1,900 feet of the diffusers as specified in the Coastal Commission Permit of February 28, 1974, Page 3, Item C.

APPENDIX F

EVACUATION MODEL

APPENDIX F

EVACUATION MODEL

"Evacuation," used in the context of offsite emergency response in the event of substantial amount of radioactivity release to the atmosphere in a reactor accident, denotes an early and expeditious movement of people to avoid exposure to the passing radioactive cloud and/or to acute ground contamination in the wake of the cloud passage. It should be distinguished from "relocation" which denotes a post-accident response to reduce exposure from long-term ground contamination. The Reactor Safety Study[1] (RSS) consequence model contains provision for incorporating radiological consequence reduction benefits of public evacuation. Benefits of a properly planned and expeditiously carried out public evacuation would be well manifested in reduction of acute health effects associated with early exposure; namely, in the number of cases of acute fatality and acute radiation sickness which would require hospitalization. The evacuation model originally used in the RSS consequence model is described in WASH-1400[1] as well as in NUREG-0340.[2] However, the evacuation model which has been used herein is a modified version[3] of the RSS model and is, to a certain extent, site-emergency-planning oriented. The modified version is briefly discussed below.

The model utilizes a circular area with a specified radius (such as a 16-km (10-mi) plume exposure pathway emergency planning Zone (EPZ)), with the reactor at the center. It is assumed that people living within portions of this area would evacuate if an accident should occur involving imminent or actual release of significant quantities of radioactivity to the atmosphere.

Significant atmospheric releases of radioactivity would in general be preceded by one or more hours of warning time (postulated as the time interval between the awareness of impending core melt and the beginning of the release of radioactivity from the containment building). For the purpose of calculation of radiological exposure, the model assumes that all people who live in a fan-shaped area (fanning out from the reactor) within the circular zone, with the downwind direction as its median (i.e., those people who would potentially be under the radioactive cloud that would develop following the release) would leave their residences after a specified amount of delay time* and then evacuate. The delay time is reckoned from the beginning of the warning time and is the sum of the time required by the reactor operators to notify the responsible authorities; the time required by the authorities to interpret the data, decide to evacuate, and direct the people to evacuate; and the time required for the people to mobilize and get underway.

The model assumes that each evacuee would move radially outward in the downwind direction with an average effective speed* (obtained by dividing the zone radius by the average time taken to clear the zone after the delay time), over a fixed distance* from the evacuee's starting point, which is somewhat greater than the zone radius. This distance is selected to be 24 km (15 mi) when the selected zone radius is 16 km (10 mi). After reaching the end of the travel distance the evacuee is assumed to receive no further radiation exposure. Persons who are outside the evacuation radius are assumed to remain in place for seven days prior to relocating, unless remaining for that long a period of time would produce a dose greater than 200 rem to the whole body. In that case, relocation takes place after 24 hours, with a dose appropriate to that time period.

The model incorporates a finite length of the radioactive cloud in the downwind direction, which would be determined by the product of the duration over which the atmospheric release would take place and the average windspeed during the release. It is assumed that the front and the back of the cloud formed would move with an equal speed, which would be the same as the prevailing windspeed; therefore, its length would remain constant at its initial value. At any time after the release, the concentration of radioactivity is assumed to be uniform over the length of the cloud. If the delay time were less than the warning time, then all evacuees would have a headstart, i.e., the cloud would be trailing behind the evacuees initially. On the other hand, if the delay time were more than the warning time, then depending on initial locations of the evacuees, there are possibilities that (a) an evacuee will still have a headstart, (b) the cloud would be already overhead when an evacuee starts to leave, or (c) an evacuee would be initially trailing behind the cloud. However, this initial picture of cloud-people disposition would change as the evacuees travel, depending on the

*Assumed to be constant value for all evacuees.

relative speed and position between the cloud and the people. The cloud and an evacuee might overtake one another one or more times before the evacuee would reach his or her destination. In the model, the radial position of an evacuating person, while stationary or in transit, is compared to the front and the back of the cloud as a function of time to determine a realistic period of exposure to airborne radionuclides. The model calculates the time periods during which people are exposed to radionuclides on the ground while they are stationary and while they are evacuating. Because radionuclides would be deposited continually from the cloud as it passed a given location, a person who is under the cloud would be exposed to ground contamination less concentrated than if the cloud had completely passed. To account for this, at least in part, the revised model assumes that persons are (a) exposed to the total ground contamination concentration which is calculated to exist after complete passage of the cloud after they are completely passed by the cloud, (b) exposed to one half the calculated concentration when anywhere under the cloud; and (c) not exposed when they are in front of the cloud. The model provides for use of different values of the shielding protection factors for exposure due to airborne radioactivity and contaminated ground. Breathing rates for stationary and moving evacuees during delay and transit periods are specifically included.

It is realistic to expect that authorities would evacuate persons at distances from the site where exposures above the threshold for causing acute fatality could occur, regardless of the EPZ distance. Figure F-1 illustrates the reduction in acute fatalities that can occur by extending evacuation to distances up to 48 km (30 mi) from the San Onofre site. (The evacuation distance used in the Reactor Safety Study[1] was 40 km (25 mi).) Also illustrated in Figure F-1 is a more pessimistic case for which no early evacuation is assumed. For this case, all persons within 16 km (10 mi) of the plant are assumed to be exposed for the first 24 hours following an accident and are then relocated. Compared to the pessimistic scenario, evacuation of a 48 km (30-mi) zone shows a reduction in acute fatalities of a factor of 10 at 10^{-8} probability.

The model has the same provision for calculation of the economic cost associated with implementation of evacuation as in the original RSS model. For this purpose, the model assumes that for atmospheric releases lasting three hours or less, all people living within a circular area of 8-km (5 mi) radius centered at the reactor plus all people within a 45-degree angular sector within the plume exposure pathway EPZ and centered on the downwind direction will be evacuated and temporarily relocated. However, for releases exceeding three hours, the cost of evacuation is based on the assumption that all people within the plume exposure pathway EPZ would be evacuated and temporarily relocated. For either of these situations, the cost of evacuation and relocation is assumed to be $125 (1980 dollars) per person which includes cost of food, and temporary sheltering for a period of one week.

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1. "Reactor Safety Study," WASH-1400, USNRC Report NUREG-75/014, October 1975.*

2. "Overview of the Reactor Safety Study Consequences Model," USNRC Report NUREG-0340, October 1977.*

3. "A Model of Public Evacuation for Atmospheric Radiological Releases," SAND 78-0092, June 1978.**

*Available from the NRC/GPO Sales Program, Washington, DC 20555, and the National Technical Information Service, Springfield, VA 22161.
**Available for inspection and copying for a fee in the NRC Public Document Room, 1717 H St. N.W., Washington, DC 20555.

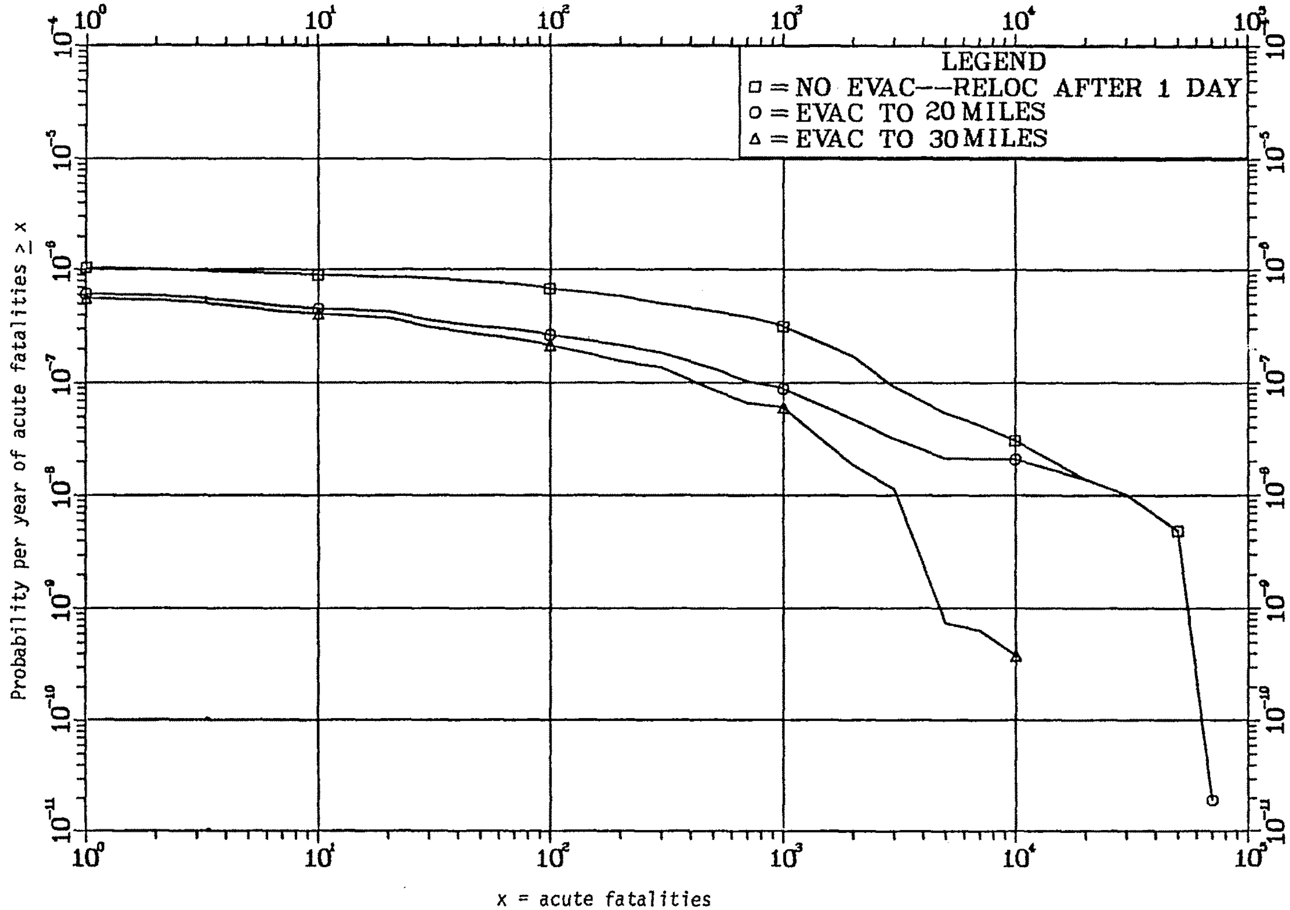

Figure F-1. Probability distribution of acute fatalities. (See Section 7.1.4.6 for discussion of uncertainties in risk estimates.)
(To change miles to kilometers, multiply by 1.6.)

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16. ABSTRACT (*200 words or less*)

A Final Environmental Statement related to the operation of San Onofre Nuclear Generating Station, Units 2 and 3 by Southern California Edison Company, et al (Docket Nos. 50-361/362), located in San Diego, California, has been prepared by the Office of Nuclear Reactor Regulation of the Nuclear Regulatory Commission. The statement reports on the staff's review of the impact of operation of the plant. Also included are comments of state and federal government agencies on the Draft Environmental Statement and its Supplement for this project and staff responses to these comments. The NRC staff has concluded, based on a weighing of environmental, technical and other factors, that operating licenses could be granted.

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NUREG-2157
Volume 1

Generic Environmental Impact Statement for Continued Storage of Spent Nuclear Fuel

Final Report

Office of Nuclear Material Safety and Safeguards

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NUREG-2157
Volume 1

Generic Environmental Impact Statement for Continued Storage of Spent Nuclear Fuel

Final Report

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Office of Nuclear Material Safety and Safeguards
U.S. Nuclear Regulatory Commission
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Abstract

This *Generic Environmental Impact Statement for Continued Storage of Spent Nuclear Fuel* (GEIS) generically determines the environmental impacts of continued storage, including those impacts identified in the remand by the Court of Appeals in the *New York v. NRC* decision, and provides a regulatory basis for a revision to 10 CFR 51.23 that addresses the environmental impacts of continued storage for use in future NRC environmental reviews. In this context, "the environmental impacts of continued storage" means those impacts that could occur as a result of the storage of spent nuclear fuel at at-reactor and away-from-reactor sites after a reactor's licensed life for operation and until a permanent repository becomes available. The GEIS evaluates potential environmental impacts to a broad range of resources. Cumulative impacts are also analyzed.

Because the timing of repository availability is uncertain, the GEIS analyzes potential environmental impacts over three possible timeframes: a short-term timeframe, which includes 60 years of continued storage after the end of a reactor's licensed life for operation; an additional 100-year timeframe (60 years plus 100 years) to address the potential for delay in repository availability; and a third, indefinite timeframe to address the possibility that a repository never becomes available. All potential impacts in each resource area are analyzed for each continued storage timeframe.

The GEIS contains several appendices that discuss specific topics of particular interest, including the technical feasibility of continued storage and repository availability as well as the two technical issues involved in the remand of *New York v. NRC*—spent fuel pool leaks and spent fuel pool fires. Finally the GEIS contains NRC's responses to public comments on the draft GEIS and proposed Rule and in doing so provides additional technical background on, and explanation of, the GEIS's analyses and conclusions.

The GEIS also discusses the NRC's Federal action—the adoption of a revised Rule, 10 CFR 51.23, to codify (i.e., adopt into regulation) the analysis in the GEIS of the environmental impacts of continued storage of spent fuel—and the options the NRC could take under the no-action alternative.

Contents

Figures

Tables

ES.3 Why Has the NRC Developed a Generic Environmental Impact Statement?

Since the Waste Confidence Rule was originally developed in 1984, the NRC has periodically updated the Rule, with the last update completed in 2010. A number of parties challenged the 2010 Waste Confidence Rule in court, and in June 2012, the Court of Appeals for the District of Columbia Circuit ruled that the 2010 Waste Confidence rulemaking did not satisfy the NRC's NEPA obligations. The Court of Appeals identified deficiencies in the 2010 Waste Confidence rule related to the NRC's environmental analysis of spent fuel pool fires and leaks, and the environmental impacts should a repository not become available.

In response to the Court of Appeals' ruling, the Commission decided that the NRC would not issue any final licenses that relied upon the Waste Confidence Rule until the NRC addressed the deficiencies identified by the Court of Appeals (Commission Order CLI–12–16). The Commission separately directed the staff to develop an updated Waste Confidence decision and Rule supported by an environmental impact statement (SRM-COMSECY-12-0016). The staff has prepared this GEIS to satisfy its NEPA obligations regarding the environmental impacts of continued storage of spent fuel in an efficient manner. The GEIS provides a regulatory basis for the revision of the Rule. Chapter 1 of the GEIS provides a more detailed discussion of the history of the Waste Confidence rulemaking.

> To comply with **The National Environmental Policy Act of 1969 (NEPA)** Federal agencies:
>
> - assess the environmental impacts of major Federal actions,
> - consider the environmental impacts in making decisions, and
> - disclose the environmental impacts to the public.

ES.4 What is the Proposed Action Being Addressed in this GEIS?

The proposed Federal action is the adoption of a revised rule—10 CFR 51.23—that codifies the analysis in the GEIS of the environmental impacts of continued storage of spent fuel.

> **Why is the NRC evaluating continued storage on a generic basis?**
>
> The NRC considers the continued storage of spent fuel an activity that is similar for all commercial nuclear power plants and storage facilities. Therefore, a generic analysis is an appropriate, effective, and efficient method of evaluating the environmental impacts of continued storage. Other examples of NRC generic environmental evaluations include the License Renewal GEIS (NUREG–1437), the Decommissioning GEIS (NUREG–0586), and the In-Situ Leach Uranium Milling Facilities GEIS (NUREG–1910).

- The *short-term storage* timeframe (60 years beyond the licensed life for operation of the reactor) includes routine maintenance and monitoring of the spent fuel pool and ISFSI and transferring spent fuel from pools to dry cask storage. Because decommissioning is required to be completed within 60 years after a reactor shuts down (unless additional time is necessary to protect public health and safety), the NRC assumes that all spent fuel will be moved from spent fuel pools to dry cask storage by the end of the short-term storage timeframe. For an away-from-reactor ISFSI, this timeframe includes construction and operation, including routine maintenance and monitoring, at the facility.
- The *long-term storage* timeframe (100 years beyond the initial 60-year [short-term] storage timeframe) includes activities such as continued facility maintenance, construction and operation of a DTS, and replacement of ISFSI and DTS facilities, including casks.
- The *indefinite storage* timeframe (no repository becomes available) assumes that the activities associated with long-term storage continue indefinitely, with ISFSI and DTS facilities being replaced at least once every 100 years.

MOX fuel is a type of nuclear reactor fuel that contains plutonium oxide mixed with either natural or depleted uranium oxide, in ceramic pellet form. This fuel differs from conventional nuclear fuel, which is made of pure uranium oxide.

Small modular reactors are nuclear power plants smaller in size (e.g., 300 MW(e)) than current generation baseload plants (e.g., 1,000 MW(e) or higher). These compactly designed reactors are factory-fabricated and can be transported by truck or rail to a nuclear power plant site.

The NRC also looked at ongoing regulatory activities that could affect the continued storage of spent fuel, including regulatory changes resulting from lessons learned from the September 11, 2001 terrorist attacks and the March 11, 2011 earthquake and tsunami that damaged the Fukushima Dai-ichi plant in Japan. Appendix B discusses a number of ongoing regulatory program reviews that ensure the safety and security of spent fuel storage and transportation.

ES.14 How did the NRC Describe Environmental Impacts?

NRC used terms from other NEPA documents, such as those for license renewal or new reactors, to define the standard of significance for assessing environmental issues.

SMALL—Environmental effects are not detectable or are so minor that they will neither destabilize nor noticeably alter any important attribute of the resource.

MODERATE—Environmental effects are sufficient to alter noticeably, but not to destabilize, important attributes of the resource.

Abstract

This *Generic Environmental Impact Statement for Continued Storage of Spent Nuclear Fuel* (GEIS) generically determines the environmental impacts of continued storage, including those impacts identified in the remand by the Court of Appeals in the *New York v. NRC* decision, and provides a regulatory basis for a revision to 10 CFR 51.23 that addresses the environmental impacts of continued storage for use in future NRC environmental reviews. In this context, "the environmental impacts of continued storage" means those impacts that could occur as a result of the storage of spent nuclear fuel at at-reactor and away-from-reactor sites after a reactor's licensed life for operation and until a permanent repository becomes available. The GEIS evaluates potential environmental impacts to a broad range of resources. Cumulative impacts are also analyzed.

Because the timing of repository availability is uncertain, the GEIS analyzes potential environmental impacts over three possible timeframes: a short-term timeframe, which includes 60 years of continued storage after the end of a reactor's licensed life for operation; an additional 100-year timeframe (60 years plus 100 years) to address the potential for delay in repository availability; and a third, indefinite timeframe to address the possibility that a repository never becomes available. All potential impacts in each resource area are analyzed for each continued storage timeframe.

The GEIS contains several appendices that discuss specific topics of particular interest, including the technical feasibility of continued storage and repository availability as well as the two technical issues involved in the remand of *New York v. NRC*—spent fuel pool leaks and spent fuel pool fires. Finally the GEIS contains NRC's responses to public comments on the draft GEIS and proposed Rule and in doing so provides additional technical background on, and explanation of, the GEIS's analyses and conclusions.

The GEIS also discusses the NRC's Federal action—the adoption of a revised Rule, 10 CFR 51.23, to codify (i.e., adopt into regulation) the analysis in the GEIS of the environmental impacts of continued storage of spent fuel—and the options the NRC could take under the no-action alternative.

future licensing reviews that are the bases for the cost analysis, while the NRC addresses unquantified costs and benefits throughout Chapter 7.

Section 7.6 summarizes and compares the estimated costs and benefits of the proposed action and the potential options in the case of no action. The cost for the proposed action (adopting a revised 10 CFR 51.23) is significantly lower than the cost for any of the no-action options. This occurs primarily because the NRC does not undertake site-specific reviews of the continued storage issue in the course of individual licensing proceedings as part of the proposed action. In general, the potential options in the case of no action are more costly than the proposed action.

The NRC provides cost information about continued storage facilities and activities in Chapter 2 in response to a large number of public comments on the draft GEIS that requested this information.

ES.21 What is the NRC's Recommendation?

Section 7.7 of the GEIS provides NRC's recommendation that the proposed action is the preferred alternative. The NRC recommendation is based on (1) the NRC's analysis of the cost-benefit balance of the proposed action and the options in the case of no action as presented in Chapter 7; (2) the NRC's consideration of public-scoping and draft GEIS comments in the development of the final GEIS; (3) the lack of environmental impacts associated with either the proposed action or the NRC's options in the case of no action; and (4) the determination that the environmental impacts of continued storage analyzed in the GEIS are unaffected by the NRC's choice of a particular administrative approach for considering the environmental impacts of continued storage in NRC licensing processes.

> The NRC recommendation is to select the proposed action—adopting a revision to 10 CFR 51.23 that codifies the impact determinations from the GEIS—as the preferred alternative.

ES.22 How is the GEIS Related to the Rule?

This GEIS provides a regulatory basis for the NRC's revised Rule, 10 CFR 51.23. Appendix B of the GEIS contains detailed information about the previous Waste Confidence proceedings, and addresses two relevant topics from Waste Confidence proceedings: (1) the technical feasibility of continued safe storage and (2) repository availability. NRC's conclusions regarding these topics continue to undergird the agency's environmental analysis.

required to complete decommissioning within 60 years after the permanent cessation of operation. Consistent with this requirement, the NRC assumes that, by the end of the short-term storage timeframe, a licensee will either terminate its Part 50 or Part 52 license and receive a specific Part 72 ISFSI license (see 10 CFR Part 72, Subpart C) or apply to receive Commission approval under 10 CFR 50.82(a)(3) or 52.110(c) to continue decommissioning under its Part 50 or Part 52 license. Accordingly, the NRC would conduct any appropriate site-specific NEPA analysis for either issuance of a Part 72 ISFSI license upon termination of the licensee's Part 50 or Part 52 license or approval to continue decommissioning beyond 60 years after ceasing operations in accordance with 10 CFR 50.82(a)(3) or 52.110(c). Further, the NRC assumes that replacing an ISFSI and licensing a DTS are licensing actions that would be subject to separate site-specific NEPA reviews. The ISFSI and DTS would be decommissioned separately.

- Construction, operation, and replacement of the DTS are assumed to occur within the long-term storage timeframe. If the DTS is built at the beginning of the long-term storage timeframe, it could be near the end of its useful life by the end of that storage timeframe. To be conservative, the NRC included the impacts of replacing the DTS one time during the long-term storage timeframe.
- Because an away-from-reactor ISFSI could store fuel from several different reactors, the earliest an away-from-reactor ISFSI would enter the short-term timeframe is when the first of these reactors reaches the end of its licensed life for operation.
- The amount of spent fuel generated is based on the assumption that the nuclear power plant operates for 80 years (40-year initial term plus two 20-year renewed terms).[4]
- A typical spent fuel pool of 700 metric tons of uranium storage capacity reaches its licensed capacity limit about 35 years into the licensed life for operation of a reactor. At that point, some of the spent fuel would need to be removed from the spent fuel pool and transferred to a dry cask storage system at either an at-reactor or away-from-reactor ISFSI.
- The environmental impacts of constructing a "spent fuel pool island," which allows the spent fuel pool to be isolated from other reactor plant systems to facilitate decommissioning, are considered within the analysis of cumulative effects in Chapter 6. Because a new spent fuel pool cooling system would be smaller in size and have fewer associated impacts than existing spent fuel pool cooling systems, the environmental impacts of operating the new spent fuel pool cooling system in support of continued storage in the spent fuel pool, would be bound by the impacts of operating the existing cooling system described in Chapter 4.
- It is assumed that an ISFSI of sufficient size to hold all spent fuel generated will be constructed during the licensed life for operation.

[4] The Commission's regulations provide that renewed operating licenses may be subsequently renewed, although no licensee has yet submitted an application for such a subsequent renewal. This GEIS included two renewals as a conservative assumption in evaluating potential environmental impacts.

Because continued operation of the ISFSI, construction and operation of the DTS, and replacement of the ISFSIs and DTSs would not significantly alter the landscape of an at-reactor ISFSI, the NRC concludes that the potential environmental impacts on aesthetic resources during long-term storage would be SMALL.

4.14.3 Indefinite Storage

If a repository is not available, current practices of using at-reactor ISFSIs are expected to continue indefinitely. At the end of each 100-year cycle, the previously reclaimed land would be used to construct the replacement ISFSIs and DTSs. The potential activities and their impacts would be the same as those described in Section 4.14.2 for long-term storage, but would continue to occur repeatedly. Therefore, the NRC concludes that the indefinite onsite storage of spent fuel would result in SMALL impacts on aesthetic resources.

4.15 Waste Management

This section describes potential environmental impacts from low-level radioactive waste (LLW), mixed waste, and nonradioactive waste management and disposal caused by the continued storage of spent fuel in spent fuel pools and at-reactor ISFSIs.

Section 3.14 identified the types of waste generated by continued storage of spent fuel, including LLW, mixed waste, hazardous waste, and nonradioactive, nonhazardous waste. The environmental impacts of hazardous waste and nonradioactive, nonhazardous waste are discussed together in this section as nonradioactive waste, unless otherwise noted.

Impacts from the transportation of waste are discussed in Section 4.16. The public and occupational health impacts associated with at-reactor radioactive waste-management activities at nuclear plants are addressed in Section 4.17.

4.15.1 Short-Term Storage

The impacts associated with the management and disposal of LLW, mixed waste, and nonradioactive waste during short-term continued storage are discussed in the following sections.

4.15.1.1 Low-Level Radioactive Waste

The continued operation of a spent fuel pool would generate minimal amounts of LLW such as wet wastes from processing and recycling contaminated liquids. In the License Renewal GEIS, the environmental impacts associated with the management, onsite storage, and disposal of LLW for an additional 20 years of operation were determined to be SMALL during normal reactor operation (NRC 2013a). The NRC concluded impacts from LLW would be SMALL

2001). The NRC assumes that an away-from-reactor ISFSI at any site has the same spent fuel capacity and a similar physical size; therefore, doses to workers and to the public would be similar to those calculated for the PFSF. The NRC concludes that public and occupational health impacts from operations during the long-term storage timeframe would be minor.

During the long-term storage timeframe, the NRC expects that the licensee would have to build a DTS for repackaging of spent fuel canisters. The operation of the DTS would involve increased doses to workers and a very small increase in dose levels at the site boundary (estimated at roughly 0.8 km [0.5 mi] based on the size of the site). However, the licensee would still be required to comply with the dose limits established by 10 CFR Part 72 and 10 CFR Part 20. In addition, the NRC assumes that the casks, pads, canister transfer building, and DTS would require replacement during the long-term storage timeframe. The health impacts related to these activities would be similar to those for the original construction of the facility.

Based on the information above, the NRC concludes that the public and occupational health impacts of ISFSI operations and construction and demolition activities during the long-term timeframe of storage would be SMALL.

5.17.3 Indefinite Storage

The public and occupational impacts of continuing to store spent fuel without a repository would be similar to those described in Section 5.17.2. The types of activities (operation, maintenance, and replacement) and associated human health impacts would remain the same. The main difference is that these activities would be repeated over a longer period of time. Based on this information, the NRC concludes that the impacts on human health during long-term storage at an away-from-reactor ISFSI would be SMALL.

5.18 Environmental Impacts of Postulated Accidents

In this section, the NRC considers the environmental impacts of postulated accidents involving continued storage of spent fuel at an away-from-reactor ISFSI. The fuel will be stored in dry storage casks licensed by the NRC. As discussed in Chapter 1, the NRC assumes that a DTS would be constructed to facilitate canister and cask replacement for long-term and indefinite storage. The consequences of accidents for a dry cask storage facility are summarized in Sections 4.18.1.2 and 4.18.2.2. The types and consequences of accidents for the away-from-reactor ISFSI are represented by the Chapter 4 results because of the similarities between the at-reactor ISFSIs and any away-from-reactor ISFSI (i.e., because the types of casks used to store the fuel and the process for licensing those casks are the same).

This section of the GEIS follows a different format than the rest of the document. Because the impacts from accidents are substantially the same across the three timeframes—short-term, long-term, and indefinite—the GEIS presents the various accident types only once.

NRC regulations at 10 CFR Part 72, "Licensing Requirements for the Independent Storage of Spent Nuclear Fuel, High-Level Radioactive Waste, and Reactor-Related Greater than Class C Waste," require that structures, systems, and components important to safety shall be designed to withstand the effects of natural phenomena (such as, earthquakes, tornadoes, hurricanes) and human-induced events without loss of capability to perform their safety functions. NRC siting regulations at 10 CFR Part 72, Subpart E, "Siting Evaluation Factors," also require applicants to consider, among other things, physical characteristics of sites that are necessary for safety analysis or that may have an impact on plant design (e.g., the design earthquake). These characteristics are to be identified and characterized so that they may be taken into consideration when determining the acceptability of the site and design criteria of the facility.

In the PFSF EIS, the NRC examined environmental impacts from accidents at the proposed PFSF. This included two events (i.e., extreme winds and 100 percent air duct blockage) that could cause higher-than-normal radiation exposures to workers. In that analysis, the NRC postulated that the high-wind event resulted in wind-borne missiles that damaged the concrete overpack, which resulted in reduced shielding. The reduced shielding would cause slightly higher occupational doses and only negligible increases in radiation doses to a member of the public at the boundary of the owner-controlled area. The NRC considered the occupational doses that would be received upon transfer of the undamaged canister to a replacement cask. The NRC estimated that the dose from transfer operations would result in a collective occupational dose of 2.47 person-mSv (247 person-mrem). In the second event involving blocked vents, the NRC estimated that the dose to a worker that removes the blockage from the vents would be 0.586 mSv (58.6 mrem) to the hands and forearms, and 0.386 mSv (38.6 mrem) to the chest, which is below regulatory limits for workers (NRC 2001). Because of the similarities between the PFSF and any away-from-reactor ISFSI (i.e., because the types of casks used to store the fuel and the process for licensing those casks are the same), the results would be similar to those for the PFSF. Therefore, the impacts of these accidents would be minor.

In addition to the credible events described above, for the PFSF the NRC also considered an accident, not considered credible, in which a canister leaks. The NRC estimated that the resulting total effective dose equivalent resulting from a 30-day leak to an individual at the owner-controlled area boundary was 0.76 mSv (76 mrem). Radiation doses after the first 30 days that result from radioactive material deposited on the ground were 0.027 mSv/yr (2.7 mrem/yr) (NRC 2001). These values are below dose limits in 10 CFR Part 20 and 10 CFR 72.106. As a result, the NRC determined that these impacts would have been SMALL (NRC 2001). Because of the similarities between the facilities, the results would be similar for any away-from-reactor ISFSI and the impacts would be minor.

While the results described from the PFSF EIS are specific to that facility, the PFSF and away-from-reactor ISFSI are similar and subject to the same regulations for casks and operations.

The NRC therefore concludes that these results are representative of the impacts for an away-from-reactor ISFSI at a different location. Therefore, the NRC concludes that the impacts of postulated accidents would be SMALL during the three storage timeframes.

5.19 Potential Acts of Sabotage or Terrorism

Section 4.19 provides background regarding the NRC approach to addressing acts of terrorism in relation to dry cask storage. That information is also applicable to an away-from-reactor ISFSI. As with the accident impacts analysis in Section 5.18, the impacts from terrorist acts are substantially the same across the three timeframes—short-term, long-term, and indefinite—and are therefore discussed only once.

The same safeguards regulations (10 CFR Part 72, Subpart H) apply to both an at-reactor ISFSI under a site-specific license and an away-from-reactor ISFSI. Safeguard requirements at at-reactor specifically licensed ISFSIs are described in Section 4.19.2 of this GEIS. In that section, the NRC concluded that both the probability and consequences of a successful attack on an at-reactor ISFSI are low and, therefore, the environmental risk is SMALL. Therefore, the NRC concludes that the results from Section 4.19.2 would also be applicable to an away-from-reactor ISFSI, and the associated impacts would be SMALL during the three storage timeframes.

5.20 Summary

The impact levels determined by the NRC in the previous sections for away-from-reactor dry cask storage of spent fuel are summarized in Table 5-1. For most impact areas, the impact levels are denoted as SMALL, MODERATE, and LARGE as a measure of their expected adverse environmental impacts. In other impact areas, the impact levels are denoted according to the types of findings required under applicable regulatory or statutory schemes (e.g., “disproportionately high and adverse” for environmental justice impacts).

For a number of the resource areas, the impact determinations for all three timeframes are SMALL. For air quality and terrestrial ecology, there is the potential for a MODERATE impact during the construction of the ISFSI. For environmental justice, special status species and habitats, and historic and cultural resources, the results are highly site-specific. While it is possible the ISFSI could be built and operated with no noticeable impacts on these resources, a definitive conclusion cannot be drawn in this GEIS. For socioeconomics (taxes), aesthetics, and traffic, there are impacts that could be greater than SMALL that will continue throughout the existence of the ISFSI. The tax impacts are beneficial in nature. Finally, there is the potential for a MODERATE impact from the disposal of nonradioactive waste in the indefinite timeframe if that waste exceeds the capacity of nearby landfills.

6.4.16.2 Potential Cumulative Impacts from Other NRC-Regulated or Spent Fuel-Related Activities

Cumulative impacts on public and occupational health could result from other NRC-regulated or spent fuel-related activities, such as reactor plant shutdown activities prior to decommissioning, decommissioning activities, construction of infrastructure to support away-from-reactor ISFSIs, and preparation activities to enable transportation of waste to a repository. The NRC has evaluated environmental impacts from these activities in the Decommissioning GEIS (NRC 2002) for reactor decommissioning and the PFSF EIS (NRC 2001a) for ISFSI decommissioning and found the public and occupational health impacts to be SMALL. The NRC also evaluated environmental impacts from infrastructure to support away-from-reactor ISFSIs in the PFSF EIS (NRC 2001a) and found the public and occupational health impacts to be SMALL. For activities related to spent fuel transportation to a repository, such as spent fuel storage maintenance activities that involve bare fuel handling in a postulated DTS at nearby facilities, as noted in Sections 4.17 and 5.17, the public and occupational health impacts would be SMALL and would not aggregate to more significant impacts, given the limited number of facilities within 80 km (50 mi) expected to be in the decommissioning phase of their lifecycle.

6.4.16.3 Conclusion

Cumulative impacts on public and occupational health include the incremental effects from continued storage when added to the aggregate effects of other past, present, and reasonably foreseeable future actions. As described in Sections 4.17 and 5.17, the incremental impacts from continued storage on public and occupational health is SMALL for all timeframes at both at-reactor and away-from-reactor storage facilities. The cumulative impacts from continued storage when added to other past, present, and reasonably foreseeable Federal and non-Federal activities are expected to be SMALL because storage facilities, reactors, and other proposed industrial buildings would be required to meet regulations such as the Occupational Safety and Health Administration's General Industry Standards (29 CFR Part 1910) and Construction Industry Standards (29 CFR Part 1926) and, as applicable, operated under NRC regulations such as 10 CFR Part 72 and 10 CFR Part 20.

6.4.17 Environmental Impacts of Postulated Accidents

This section evaluates the effects of continued storage on accident risk when added to the aggregate effects of other past, present, and reasonably foreseeable future actions. As described in Sections 4.18 and 5.18, the incremental impacts from continued storage on environmental impacts of postulated accidents is SMALL for all timeframes at both at-reactor and away-from-reactor storage facilities.

The geographic area considered in the cumulative accident risk assessment is an 80-km (50-mi) radius from an at-reactor or away-from-reactor storage facility. The cumulative analysis

considers risk from potential accidents from other nuclear plants or storage facilities that have the potential to increase risks at any location within 80 km (50 mi) of the shutdown reactor or storage facility. It is possible that one or more other types of nuclear facilities that support the nuclear fuel cycle may be located within an 80-km (50-mi) radius, but these facilities generally involve very low accident risk (51 FR 30028). Therefore, the analysis below focuses on the cumulative risk from reactors and storage facilities.

6.4.17.1 Potential Cumulative Impacts from General Trends and Activities

Based on a review of the other activities that can occur near proposed new at-reactor storage facilities, there are two scales of cumulative impacts on accident risk, including (1) cumulative impacts due to the various impacts from an individual power plant and storage facility over time (e.g., annual design basis and severe accident risks at a reactor), and (2) cumulative impacts due to closely sited operating or decommissioning reactors (e.g., design basis and severe accident risks at other reactors located within 80 km [50-mi]) or other radioactive facilities. In addition, climate change can impact accident risk due to higher or lower intensity or frequency of natural phenomena hazards (e.g., precipitation, tornadoes, hurricanes) that could result in radiological accidents.

The magnitude of cumulative accident impacts resulting from all general trends taking place within the 80-km (50-mi) region of a power plant and storage facility would likely be limited because:

1. Estimates of average individual early fatality and latent cancer fatality risks are well below the Commission’s safety goals at all plants (51 FR 30028).
2. The Commission has determined that the probability-weighted consequences of severe accidents of a nuclear power plant are SMALL (10 CFR Part 51, Appendix B, Table B-1).
3. The severe accident risk due to any particular nuclear power plant gets smaller as the distance from that plant increases. However, the combined risk at any location within 80 km (50 mi) of a reactor site would be bounded by the sum of risks for all of these operating and proposed nuclear power plants. Even though several plants and other nuclear facilities could potentially be included in the combination, this combined risk would still be low.

Because design basis accidents at nearby power plants and storage facilities are individually unlikely to occur more than once over the life of a facility, and licensees must show that accident consequences of design basis accidents are mitigated to acceptable levels of dose offsite, the cumulative impact of design basis accidents is very small. Based on the above discussion, the NRC concluded that, in all new reactor EISs published through February 2013 (e.g., NRC 2011a–e, 2013c), the cumulative risks from design basis and severe accidents at any location within 80 km (50 mi) of a reactor would be SMALL.

Appendix B

Technical Feasibility of Continued Storage and Repository Availability

B.1 Introduction

In this *Generic Environmental Impact Statement for Continued Storage of Spent Nuclear Fuel* (GEIS), the U.S. Nuclear Regulatory Commission (NRC) addresses the environmental impacts of continuing to store spent nuclear fuel (spent fuel) at a reactor site or at an away-from-reactor storage facility, after the end of a reactor's licensed life for operation until final disposition in a geologic repository ("continued storage"). This GEIS provides a regulatory basis for the NRC's proposed amendment to Title 10 of the *Code of Federal Regulations* (CFR) Part 51. Historically, past Waste Confidence proceedings included a Decision with five findings that addressed technical feasibility of a mined geologic repository, the degree of assurance that disposal would be available by a certain time, and the degree of assurance that spent fuel and commercial high-level waste could be managed safely without significant environmental impacts for a certain period beyond the expiration of plants' operating licenses. Preparation of and reliance upon a GEIS is a fundamental departure from the approach used in past Waste Confidence proceedings. This GEIS acknowledges the uncertainties in the Commission's prediction of repository availability and provides an environmental analysis of three possible storage timeframes. To this end the GEIS considers impacts for three possible timeframes constrained by repository availability, including the impacts from indefinite storage, should a repository never become available.

The NRC's underlying conclusions regarding the technical feasibility of continued storage and a repository continue to undergird its environmental analyses. These underlying conclusions, which are relevant to an analysis of the potential environmental impacts assessed in this GEIS, are discussed as two broad issues in this appendix: the NRC's technical information regarding the availability of a repository for disposal of spent fuel generated in a power reactor (Section B.2) and the technical feasibility of safe storage of spent fuel in an at-reactor or away-from-reactor storage facility until sufficient repository capacity becomes available (Section B.3). These two broad issues were addressed in the five findings contained in the Waste Confidence Decision from past Waste Confidence proceedings; this appendix addresses these issues under two broad topic areas rather than five findings. Section B.4 provides a summary of the conclusions reached in this appendix.

B.2 Repository will be Available to Dispose of Spent Fuel

Based on the analysis below and elsewhere in this GEIS, the NRC believes that the most-likely scenario is that a repository will become available to dispose of spent fuel by the end of the short-term timeframe (within 60 years of the end of a reactor's licensed life for operation). The NRC's belief is based on the resolution of two questions: whether a repository is technically feasible and, if so, how long will it take to site, license, construct, and open a repository. "Technical feasibility" simply means whether a geologic repository is technically possible using existing technology (i.e., without any fundamental breakthroughs in science and technology). If technically feasible, then the question becomes what is a reasonable timeframe for the siting, licensing, construction, and opening of a geologic repository. Both questions are discussed in detail below in Sections B.2.1 (Technical Feasibility of a Repository) and B.2.2 (Availability of a Repository).

B.2.1 Technical Feasibility of a Repository

The Commission has consistently determined that current knowledge and technology support the technical feasibility of deep geologic disposal. In its original 1984 Waste Confidence proceeding, the NRC stated that "[t]he Commission finds that safe disposal of [high-level radioactive waste and spent nuclear fuel] is technically *possible* and that it is achievable using *existing* technology" (49 FR 34658) (emphasis added). The Commission then stated: "Although a repository has not yet been constructed and its safety and environmental acceptability demonstrated, no fundamental breakthrough in science or technology is needed to implement a successful waste disposal program." Although the Commission has conducted Waste Confidence proceedings since 1984, this focal point—whether a fundamental breakthrough in science or technology is needed—continues to guide the Commission's consideration of the feasibility of spent fuel disposal. Since 1984, the technical feasibility of a geological repository has moved significantly beyond a theoretical concept.

Today, the consensus within the scientific and technical community engaged in nuclear waste management is that safe geologic disposal is achievable with currently available technology (see, e.g., Blue Ribbon Commission on America's Nuclear Future [BRC 2012], Section 4.3). Currently, 25 countries, including the United States, are considering disposal of spent or reprocessed nuclear fuel in deep geologic repositories. Repository programs in other countries, which continue to provide additional information useful to the U.S. program, are actively considering crystalline rock, clay, and salt formations as repository host media (IAEA 2005). Many of these programs have researched these geologic media for several decades.

Ongoing research in both the United States and other countries supports a conclusion that geological disposal remains technically feasible and that acceptable sites can be identified. After decades of research into various geological media, no insurmountable technical or scientific problem has emerged to challenge the conclusion that safe disposal of spent fuel and

high-level radioactive waste can be achieved in a mined geologic repository. Over the past two decades, significant progress has been made in the scientific understanding and technological development needed for geologic disposal. There is now a better understanding of the processes that affect the ability of repositories to isolate waste over long periods (e.g., the International Atomic Energy Agency's [IAEA's] *Scientific and Technical Basis for the Geologic Disposal of Radioactive Wastes, Technical Reports Series No. 413* [IAEA 2003a] and Ahn and Apted's *Geological Repository Systems for Safe Disposal of Spent Nuclear Fuels and Radioactive Wastes* [Ahn and Apted 2010]).

Further, the ability to characterize and quantitatively assess the capabilities of geologic and engineered barriers has been repeatedly demonstrated (see the Organisation for Economic Cooperation and Development, Nuclear Energy Agency's *Lessons Learnt from Ten Performance Assessment Studies* [NEA 1997]). In addition, specific sites have been investigated and extensive experience has been gained in underground engineering (see IAEA's *Radioactive Waste Management Studies and Trends, IAEA/WMDB/ST/4* [IAEA 2005] and *The Use of Scientific and Technical Results from Underground Research Laboratory Investigations for the Geologic Disposal of Radioactive Waste* [IAEA 2001]). These advances and others throughout the world (e.g., IAEA's *Joint Convention on Safety of Spent Fuel Management and on Safety of Radioactive Waste Management, INFCIRC/546* [IAEA 1997]) continue to confirm the soundness of the basic concept of deep geologic disposal (IAEA 1997). In the United States, the technical approach for safe high-level radioactive waste disposal has remained unchanged for several decades—a deep geologic repository containing natural barriers to hold canisters of high-level radioactive waste with additional engineered barriers to further retard radionuclide release. Although some elements of this technical approach have changed in response to new knowledge, safe disposal remains feasible with current technology. The recent report by the Blue Ribbon Commission on America's Nuclear Future (BRC 2012) supported geologic disposal by concluding that:

> geologic disposal in a mined repository is the most promising and technically accepted option available for safely isolating high-level nuclear wastes for very long periods of time. This view is supported by decades of expert judgment and by a broad international consensus. All other countries with spent fuel and high-level waste disposal programs are pursuing geologic disposal. The United States has many geologic media that are technically suitable for a repository.

In addition, support for the feasibility of geologic disposal can be drawn from experience gained from the review of the U.S. Department of Energy's (DOE's) license application for a high-level nuclear waste repository at Yucca Mountain, Nevada (DOE 2008a). On June 3, 2008, the DOE submitted an application for a construction authorization to the NRC, and on September 8, 2008, the NRC notified DOE that it found the application acceptable for docketing (73 FR 53284) and began its review. DOE subsequently filed a motion with an NRC Atomic Safety and Licensing Board seeking permission to withdraw the license application

(NRC 2010a). In recognition of budgetary limitations, the Commission directed the Atomic Safety and Licensing Board to complete all necessary and appropriate case management activities, and the Atomic Safety and Licensing Board suspended the proceeding. The NRC staff completed three technical review documents (i.e., NRC 2011a,b,c) covering the operational period and the postclosure period (i.e., the period after permanent closure of the repository) and one safety evaluation report on general information (NRC 2010b). The NRC staff's technical review did not identify any issues that would challenge the feasibility of geological disposal as a general matter. However, these technical reports did not include conclusions as to whether or not DOE's proposed Yucca Mountain repository satisfies the Commission's regulations and do not constitute a final judgment or determination of the acceptability of the DOE construction application.

In August 2013, the U.S. Court of Appeals for the District of Columbia Circuit (Court of Appeals) issued a writ of mandamus and directed the NRC to resume the licensing process for DOE's license application. In response, the Commission directed the NRC staff to complete and issue the safety evaluation report associated with the license application (NRC 2013). Currently, the NRC is working on completing its safety review of DOE's license application and plans to publish the remaining volumes of its safety evaluation report by January 2015.

The technical feasibility of a deep geologic repository is further supported by current DOE defense-related activities. The DOE sited and constructed, and since March 1999 has been operating, a deep geologic repository for defense-related transuranic radioactive wastes near Carlsbad, New Mexico. At this site, the DOE has successfully disposed of transuranic waste from nuclear weapons research and testing operations. This Waste Isolation Pilot Plant (WIPP) is located in the Chihuahauan Desert of southeastern New Mexico, approximately 42 km (26 mi) east of Carlsbad. The facility is used to store transuranic waste from nuclear weapons research and testing operations from past defense activities. Project facilities include mined disposal rooms 655 m (2,150 ft) underground.

The NRC recognizes the incident at WIPP on February 14, 2014, which resulted in the release of americium and plutonium from one or more transuranic (TRU) waste containers into the environment. Trace amounts of americium and plutonium are believed to have leaked through unfiltered exhaust ducts and escaped aboveground. No personnel were determined to have received external contamination; however, 21 individuals were identified through bioassay to have initially tested positive for low level amounts of internal contamination. No adverse health impacts have been reported. The DOE has issued a Phase 1 accident report on the incident (DOE 2014). Despite the event, the NRC continues to conclude that a repository is technically feasible.

In January 2013, the DOE released *Strategy for the Management and Disposal of Used Nuclear Fuel and High-Level Radioactive Waste*, a response to the Blue Ribbon Commission on America's Nuclear Future's report (DOE 2013). In this strategy document, DOE presents a

framework for "moving toward a sustainable program to deploy an integrated system capable of transporting, storing, and disposing of [spent] nuclear fuel and high-level radioactive waste from civilian nuclear power generation…" (DOE 2013). This new DOE strategy includes a nuclear waste-management system consisting of a pilot interim storage facility, a larger full-scale interim storage facility, and a geologic repository. U.S. policy remains that geologic disposal is the appropriate long-term solution for disposition of spent fuel and high-level radioactive waste.

Finally, the activities of European countries support the technical feasibility of a deep geologic repository. In late 2012, a Finnish nuclear-waste-management company (Posiva) submitted a construction license application for a geological repository for spent fuel to Finland's Radiation and Nuclear Safety Authority, and in spring 2011, Swedish nuclear authorities accepted an application from the Swedish Nuclear Fuel and Waste Management Company for permission to build a repository for spent fuel. Based on the national and international research, proposals, and experience with geologic disposal, the NRC concludes that a geologic repository continues to be technically feasible.

B.2.2 Availability of a Repository

Given the consensus that geologic repositories are technically feasible, experience to date is also relevant in determining the timeframe to successfully site, license, construct, and open a repository. Of the 24 countries other than the United States considering disposal of spent or reprocessed nuclear fuel in deep geologic repositories, 10 have established target dates for the availability of a repository.[1] The majority of the 14 countries with no established target date for repository availability rely on centralized interim storage, which may include a protracted period of at-reactor storage before shipment to a centralized facility.

While some countries have struggled with specific implementation issues, the international consensus regarding an approach to disposal in a deep geologic repository and a reasonable timeframe for a repository to become available has not been abandoned.

In 1997, the United Kingdom rejected an application for the construction of a rock characterization facility at Sellafield, leaving the country without a path forward for long-term management or disposal of intermediate-level waste or spent fuel. In 1998, an inquiry by the United Kingdom House of Lords endorsed geologic disposal but specified that public acceptance was required. As a result, the United Kingdom Government embraced a repository plan based on the principles of voluntarism and partnership between communities and

[1] The three countries with target dates that plan direct disposal of spent fuel are: Czech Republic (2050), Finland (2020), and Sweden (2025). The seven countries with target dates for disposal of reprocessed spent fuel and high-level radioactive waste are: Belgium (2035), China (2050), France (2025), Germany (2025), Japan (2030s), Netherlands (2103), and Switzerland (2042).

implementers. This led to the initiation of a national public consultation and major structural reorganization within the United Kingdom program. In 2008, the UK Government called for potential volunteers to host the repository and was expecting the repository would open around 2040 (MRWS 2012). In 2013, the Cumbria County Council voted to withdraw from the United Kingdom process to find a host community for an underground radioactive waste disposal facility and to end the site-selection process in West Cumbria. In responding to the outcome of the votes in West Cumbria, the Secretary of State for Energy and Climate Change published a Written Ministerial Statement on January 31, 2013, that made clear that the United Kingdom Government remains committed to geological disposal for the safe and secure management of higher activity radioactive waste (DECC 2013) . In July 2014, the United Kingdom continued to support geological disposal and provided a revised policy framework for implementing geological disposal that favors a voluntarist approach based on working with communities that are willing to participate in the siting process (DECC 2014). The formal process for working with communities is expected to begin in 2016.

In Germany, a large salt dome at Gorleben had been under study since 1977 as a potential spent fuel repository. After suspension of exploration in 2000, Germany resumed exploration of Gorleben as a potential spent fuel repository in 2010. In July 2013, the Site Selection Act became effective in Germany. Currently, a 33-member commission made up of representatives from societal groups, academia, and the German government is preparing proposals for site selection procedures, which are due by the end of 2015.

Initial efforts in France during the 1980s also failed to identify potential repository sites, using solely technical criteria. Failure of these attempts led to the passage of nuclear waste legislation that prescribed 15 years of research. Reports on generic disposal options in clay and granite media were prepared and reviewed by the French Nuclear Safety Authority in 2005. In 2006, the French Parliament passed new legislation designating a single site for deep geologic disposal of intermediate- and high-level radioactive waste. This facility, to be located near the town of Bure in northeastern France, is scheduled to open in 2025, about 34 years after passage of the original Nuclear Waste Law of 1991, and 19 years after site selection. On May 6, 2014, the French National Agency for Radioactive Waste Management (ANDRA) announced the actions it intends to take resulting from recent public debate on geological disposal. ANDRA announced plans for a pilot facility and improvements for greater public involvement. ANDRA anticipates completion of the license application at the end of 2017 and, subject to approvals, construction of the facility could begin in 2020 and a pilot phase could begin in 2025.

In Switzerland, after detailed site investigations in several locations, the Swiss National Cooperative for Radioactive Waste Disposal proposed, in 1993, a deep geologic repository for low- and intermediate-level waste at Wellenberg. In 1998, Swiss authorities found that technical feasibility of the disposal concept had been successfully demonstrated; however, in 2002, a public cantonal referendum rejected the proposed repository. Despite difficulties with public acceptance, Swiss authorities have gathered more than 25 years of high-quality field and

laboratory research and are anticipating constructing and operating a deep geologic repository after 2040, less than 30 years from today. A site selection plan was approved by the Federal Council in 2008 and three geological siting areas were identified by 2011 for deep geological disposal of high-level waste. A second phase is currently underway and involves regional participation and comparative studies with safety as the decision criterion.

In 1998, an independent panel reported to the Governments of Canada and Ontario on its review of Atomic Energy of Canada Ltd.'s concept of geologic disposal (CEAA 1998). The panel concluded that broad public support is necessary in Canada to ensure the acceptability of a concept for managing spent fuel. The panel also found that technical safety is a key part, but only one part, of acceptability. To be considered acceptable in Canada, the panel found that a concept for managing nuclear fuel wastes must (1) have broad public support; (2) be safe from a technical perspective; (3) have been developed within a sound ethical and social assessment framework; (4) have the support of Aboriginal people; (5) be selected after comparison with the risks, costs, and benefits of other options; and (6) be advanced by a stable and trustworthy proponent and overseen by a trustworthy regulator. Resulting legislation mandated a nationwide consultation process and widespread organizational reform.

In 2007, the Government of Canada announced its selection of the Adaptive Phased Management approach and directed the Nuclear Waste Management Organization to take at least 2 years to develop a "collaborative community-driven site-selection process." The Nuclear Waste Management Organization is using this process to open consultations with citizens, communities, Aboriginals, and other interested parties to find a suitable site in a willing host community. Nuclear Waste Management Organization's site-selection process was initiated in May 2010. For financial planning and cost estimation purposes only, the Nuclear Waste Management Organization assumes the availability of a deep geological repository in 2035, 27 years after initiating development of new site-selection criteria, 30 years after embarking on a national public consultation, and 37 years after rejection of the original geologic disposal concept (NWMO 2008). At the end of 2012, 21 communities had expressed interest in learning more about the project (NWMO 2013). As of June 2014, 14 of the initial 21 communities are still actively engaged in the siting process. In particular, four communities are continuing with more detailed analyses having completed preliminary assessments; 10 communities are still in the preliminary assessment phase; and seven communities are no longer being considered in the site selection process.

Repository development programs in Finland and Sweden are further along than in other countries but have taken time to build support from potential host communities. In Finland, preliminary site investigations started in 1986, and detailed characterizations of four locations were performed between 1993 and 2000. In 2001, the Finnish Parliament ratified the government's decision to proceed with a repository project at a chosen site only after the 1999 approval by the municipal council of the host community. In December 2012, Posiva (i.e., the nuclear-waste-management company in Finland) submitted a construction permit application for

a final repository that will hold spent fuel from Finland's nuclear reactors. In June 2014, the Radiation and Nuclear Safety Authority (STUK) in Finland estimated that it can complete its safety assessment report for the construction permit application in January 2015. Finland expects this facility to begin receipt of spent fuel for disposal in 2020, 34 years after the start of preliminary site investigations.

Between 1993 and 2000, Sweden conducted feasibility studies in eight municipalities. One site was found technically unsuitable, and two sites were eliminated by municipal referenda. Three of the remaining five sites were selected for detailed site investigations. Municipalities adjacent to two of these sites agreed to be potential hosts, and one refused. Since 2007, detailed site investigations were conducted at Östhammar and Oskarshamn, both of which already host nuclear power stations. On June 3, 2009, the Swedish Nuclear Fuel and Waste Management Company (SKB) selected the Forsmark site located in the Östhammar municipality for the Swedish spent fuel repository and, in spring 2011, SKB submitted a license application. At the request of the Swedish government, the Nuclear Energy Agency organized an international team to review the SKB license application. In June 2012, the international review team completed its review and report stating: "SKB's post-closure radiological safety analysis report, SR-Site, is sufficient and credible for the licensing decision at hand. SKB's spent fuel disposal programme is a mature programme—at the same time innovative and implementing best practice—capable in principle to fulfil the industrial and safety-related requirements that will be relevant for the next licensing steps" (NEA 2012). In April 2014, the Swedish Radiation Safety Authority, as part of its review process, circulated the license application for comment to other public authorities and environmental organizations. A government decision is expected in 2015. If Swedish authorities authorize construction, the repository could be available for disposal around 2025, about 30 years after feasibility studies began.

In the United States, the DOE is the agency responsible for carrying out the national policy to site and build a repository, which includes designing, constructing, operating, and decommissioning the repository. The time DOE will need to develop a repository site will depend upon a variety of factors, including Congressional action and funding. Public acceptance will also influence the time it will take to implement geologic disposal. The NRC, by contrast, is the agency responsible for reviewing, licensing, and overseeing the construction and operation of the repository.

In 2012, the Blue Ribbon Commission on America's Nuclear Future recommended "prompt efforts to develop one or more geologic disposal facilities" (BRC 2012). In response to the Blue Ribbon Commission's report, the DOE (2013) stated that its "…goal is to have a repository sited by 2026; the site characterized, and the repository designed and licensed by 2042; and the repository constructed and its operations started by 2048." Based on the evaluation of international experience with geologic repository programs—including the issues some countries have overcome—and the affirmation by the Blue Ribbon Commission of the geologic repository approach, the NRC continues to believe that 25 to 35 years is a reasonable period for

repository development (i.e., candidate site selection and characterization, final site selection, licensing review, and initial construction for acceptance of waste).

Although the NRC believes that 25 to 35 years is a reasonable timeframe for repository development, it acknowledges that there is sufficient uncertainty in this estimate that the possibility that more time will be needed cannot be ruled out. International and domestic experience have made it clear that technical knowledge and experience alone are not sufficient to bring about the broad social and political acceptance needed to construct a repository. The time needed to develop a societal and political consensus for a repository could add to the time to site and license a repository or overlap it to some degree.

Because the availability of a repository can be substantially affected by whatever process is employed to achieve a national consensus on repository site selection, and consistent with the decision of the Court of Appeals in *New York v. NRC*, this GEIS offers three timeframes for continued storage that reflect significant differences in the availability of the repository. The short-term timeframe assumes a repository is available 60 years after the end of a reactor's licensed life for operation. The long-term timeframe assumes a repository is not available for an additional 100 years beyond the short-term timeframe, which means a repository would be available 160 years after the end of a reactor's licensed life for operation. In recognition of the uncertainty in reaching a national consensus on repository site selection, the third timeframe assumes that a repository does not become available and the spent fuel continues to be stored indefinitely.

In the 2010 Waste Confidence decision, the Commission assessed the length of time that would be needed to site, license, construct, and open a repository. This analysis moved away from the Commission's historical practice of specifying a "target date" and instead concluded that a repository would be available "when necessary." The Commission's reluctance to select a target date was not indicative of an inability to predict the length of the process for siting, constructing, licensing, and opening a repository, but rather that identification of a specific year as a starting point was uncertain. In sum, based on experience in licensing similarly complex facilities in the United States and national and international experience with repositories already in progress, the NRC concludes a reasonable period of time for the development of a repository is approximately 25 to 35 years.

B.3 Technical Feasibility of Safe Storage

Spent fuel removed from a reactor is initially placed in a spent fuel pool for cooling. After several years (about 5 years for low-burnup fuel and up to 20 years for high-burnup fuel), the spent fuel is sufficiently cooled that it can be placed in dry cask storage assuming current

storage configurations and heat loads.[2] After the end of a reactor's licensed life for operations, spent fuel is stored in onsite spent fuel pools or in an at-reactor or away-from-reactor dry cask storage system.

Continued storage of spent fuel at at-reactor or away-from-reactor sites will be necessary until a repository is available for permanent disposal. The storage of spent fuel in any combination of storage (spent fuel pools or dry casks) will continue as a licensed activity under regulatory controls and oversight. Nonetheless, the conclusions reached by the NRC in this GEIS regarding the technical feasibility of continued storage do not rely solely on NRC's regulatory framework governing these activities. Rather, these conclusions are also based on NRC's experience with the actual storage of spent fuel under this regulatory framework and the continued application of proven fuel storage methodologies. Continued safe storage of spent fuel requires both the technical feasibility of storage methods and a regulatory framework that provides for monitoring and oversight to address the potential for evolving issues. The technical feasibility of wet storage in spent fuel pools and dry casks is discussed separately in Sections B.3.1 and B.3.2. The regulatory framework applicable to both wet and dry storage is discussed in Section B.3.3. The continuation of the institutional controls necessary to maintain safe storage is discussed in Section B.3.4.

B.3.1 Technical Feasibility of Wet Storage

The technical feasibility of continued storage in spent fuel pools is supported by a number of technical considerations. First, the integrity of spent fuel and cladding within the benign environment of the spent fuel pool's controlled water chemistry is supported by operational experience and a number of scientific studies, some of which are summarized below. Further, the spent fuel pool's robust structural design protects against a range of natural and human-induced challenges, which are discussed in detail in the following sections and in the body of the GEIS.

B.3.1.1 Integrity of Spent Fuel and Cladding in Spent Fuel Pools

In 1984, the NRC provided information supporting the low degradation rates of spent fuel in spent fuel pools based on national and international storage experience, which at that time totaled 18 years of experience with zirconium-clad fuel[3] and 12 years of experience with stainless-steel-clad fuel (49 FR 34658). Examples of the cited information are:

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[2] Appendix I provides additional information on the characteristics, storage, and transportation of high-burnup uranium oxide (UOX) spent fuel and mixed uranium-plutonium oxide (MOX) spent fuel.
[3] In 1984, only two commercial light water reactor nuclear power plants used stainless-steel-clad fuel, whereas most used zirconium-clad fuel (49 FR 34658).

1. In *Behavior of Spent Nuclear Fuel in Water Pool Storage,* Johnson (1977) reported on corrosion studies of irradiated fuel at 20 reactor pools in the United States, finding no detectable degradation of zirconium cladding.
2. At the American Nuclear Society's Executive Conference on Spent Fuel Policy and its Implications, presented in Buford, Georgia, April 2 to 5, 1978, Johnson (1978) presented "Utility Spent Fuel Storage Experience," which reported that no degradation has been observed in commercial power reactor fuel stored in onsite pools in the United States and that extrapolation of corrosion data suggests that less than a tenth of a percent of the thickness of the zirconium clad would be corroded after 100 years.
3. In *The Long-Term Storage of Irradiated CANDU Fuel Under Water*, Walker (1979) concluded that "50 to 100 years under water should not significantly affect their [spent fuel bundles] integrity."

Almost 30 years of additional experience has been gained since the completion of the first Waste Confidence proceeding in 1984, during which time the technical basis for very slow degradation rates of spent fuel in spent fuel pools has continued to grow and now includes the wet storage of high-burnup fuel. Examples of this additional experience include the following:

1. In *Durability of Spent Nuclear Fuels and Facility Components in Wet Storage*, the IAEA (1998) summarized the durability of materials in wet storage, stating: "The zirconium alloys represent a class of materials that is highly resistant to degradation in wet storage, including some experience in aggressive waters. The only adverse experience involves Zircaloy clad metallic uranium where mechanical damage to the cladding was a prominent factor during reactor discharge, exposing the uranium metal fuel to aqueous corrosion. Otherwise, the database for the zirconium alloys supports a judgment of satisfactory wet storage in the time frame of 50 to 100 years or more."
2. In *Spent Fuel Performance Assessment and Research: Final Report of a Co-Ordinated Research Project on Spent Fuel Performance Assessment and Research (SPAR) 1997–2001*, the IAEA (2003b), while discussing spent fuel storage experience, reported on a detailed review of the degradation mechanisms of spent fuel cladding under wet storage and stated that "wet storage of spent fuel only appears to be limited by adverse pool chemistry or the deterioration of the fuel storage pool structure."
3. In *Understanding and Managing Ageing of Materials in Spent Fuel Storage Facilities*, the IAEA reported that "over more than 40 years of experience with several million LWR [light water reactor] rods, power reactor fuel with zirconium alloy cladding has had an excellent durability in wet storage" (IAEA 2006). The IAEA went on to state that "destructive and non-destructive examinations of fuel rods, visual evidence and coupon studies [IAEA 2006; pp. 11, 13, 54–58] all support resistance to aqueous corrosion. There have been no reports of fission gas evolution, indicative of cladding failure in wet storage. Rod consolidation campaigns have been conducted without any indication of storage-induced degradation.

There is a sufficient database to indicate that wet storage of fuel with zirconium alloy cladding can be extended for at least several decades."

4. In *Impact of High Burnup Uranium Oxide and Mixed Uranium-Plutonium Oxide Water Reactor Fuel on Spent Fuel Management, No. NF-T-3.8*, (IAEA 2011a) the IAEA stated that because wet storage is associated with low temperatures, the clad material property degradation is expected to be low. However, the IAEA also recognized that high-burnup uranium oxide and MOX spent fuel storage in pools will increase the heat load and potentially cause radioactive releases, which may require an upgrade of the pool facility with respect to heat removal, pool cleanup systems, and additional neutron poison material in the pool water or in storage racks. In addition, the IAEA suggested that reevaluation of criticality and regulatory aspects may also be required for high-burnup fuel.

Based on available information and operational experience, degradation of the fuel cladding occurs very slowly over time in the spent fuel pool environment. Degradation of the spent fuel should be minimal over the short-term storage timeframe. The NRC expects that only routine maintenance will be needed over the short-term storage timeframe. However, it is possible that future evaluations and experience with high-burnup fuel could identify upgrades and enhancements to pool storage that would need to be implemented in the future (see discussion on regulatory framework in Section B.3.3). Although the NRC assumes in the GEIS that the spent fuel pool will be decommissioned before the end of the short-term storage timeframe, it is not aware of any information that would call into question the technical feasibility of continued safe storage of spent fuel in spent fuel pools beyond the short-term storage timeframe.

B.3.1.2 Robust Structural Design of Spent Fuel Pools

As described in Section 2.1.2.1 of the GEIS, spent fuel pools are massive, seismically designed structures that are constructed from thick, reinforced concrete walls and slabs that vary between 0.7 and 3 m (2 and 10 ft) thick. All spent fuel pools currently in operation are lined with stainless-steel liners that vary in thickness from 6 to 13 mm (0.25 to 0.5 in.).[4] NUREG–1738 (NRC 2001), *Technical Study of Spent Fuel Pool Accident Risk at Decommissioning Nuclear Power Plants* indicates that spent fuel pool structures are designed to be seismically robust (i.e., it is expected that a seismic event with peak spectral acceleration significantly larger than that of the safe shutdown earthquake would be required to produce catastrophic failure of the structure) (NRC 2001). Further, in evaluating the seismic risk to spent fuel pools, NRC (2001) stated that "[i]n boiling-water reactor (BWR) plants, the pool structures are located in the reactor building at an elevation several stories above the ground. In pressurized-water reactor (PWR)

[4] Dresden Unit 1 and Indian Point Unit 1 have no liner plates, but neither pool is currently operating. Both plants were permanently shut down more than 20 years ago and no safety-significant degradation of their concrete pool structures has been reported. At present, no spent fuel remains in either reactor's spent fuel pool.

plants, the [spent fuel pool] structures are outside the containment structure and supported on the ground or partially embedded in the ground. The location and supporting arrangement of the pool structures affect their capacity to withstand seismic ground motion beyond their design basis. The dimensions of the pool structure are generally derived from radiation shielding considerations rather than seismic demand needs. Spent fuel structures at nuclear power plants are able to withstand loads substantially beyond those for which they were designed." In *Spent Fuel Storage Operation—Lessons Learned* (IAEA 2013), the IAEA reported that pool storage is a mature technology and the latest storage pools have come through an evolutionary process and incorporate the learning from 50+ years of operating experience.

In the initial Waste Confidence proceeding, the Commission found that the risks of major accidents at spent fuel pools resulting in offsite consequences were remote because of the secure and stable character of the spent fuel in the spent fuel pool environment and the absence of reactive phenomena that might result in dispersal of radioactive material. The Commission noted that storage pools and independent spent fuel storage installations (ISFSIs) are designed to safely withstand accidents caused by either natural or man-made phenomena (49 FR 34658). By 1990, the NRC had spent several years studying the potential for a catastrophic loss of reactor spent fuel pool water, which could cause a spent fuel fire in a dry pool. The NRC concluded that, because of the large inherent safety margins in the design and construction of a spent fuel pool, no action was needed to further reduce the risk (55 FR 38472).

On March 11, 2011, an earthquake and subsequent tsunami resulted in significant damage to the nuclear facilities at Fukushima Dai-ichi. Subsequent analysis and inspections performed by Tokyo Electric Power Company personnel determined that the spent fuel pool water levels did not drop below the top of fuel in any spent fuel pool and that no significant fuel damage occurred (INPO 2011). Appendix F contains further discussion of the Fukushima event with respect to spent fuel pools.

The NRC has continued its examination of spent fuel pool storage to ensure that adequate safety is maintained and that there are no adverse environmental effects from the storage of spent fuel in spent fuel pools. The Office of Nuclear Reactor Regulation and the former Office for Analysis and Evaluation of Operational Data independently evaluated the safety of spent fuel pool storage, and the results of these evaluations were documented in a pair of memoranda to the Commission: *Resolution of Spent Fuel Storage Pool Action Plan Issues* (NRC 1996a) and *Assessment of Spent Fuel Pool Cooling* (NRC 1996b) (later published as NUREG–1275, *Vol. 12, "Operating Experience Feedback Report: Assessment of Spent Fuel Cooling*" [NRC 1997a]). As a result of these studies, the NRC and industry identified a number of follow-up activities, which are described by the NRC in a memorandum to the Commission *Follow-up Activities on the Spent Fuel Pool Action Plan* (NRC 1997b). These evaluations subsequently became part of the investigation of Generic Safety Issue 173, *Spent Fuel Pool*

Cooling for Operating Plants, which found that the relative risk posed by loss of spent fuel cooling is low compared with the risk of events not involving the spent fuel pool (NRC 2000).

The safety and environmental effects of spent fuel pool storage were also addressed in conjunction with regulatory assessments of permanently shutdown nuclear plants and decommissioning nuclear power plants. NUREG/CR–6451, *A Safety and Regulatory Assessment of Generic BWR and PWR Permanently Shutdown Nuclear Power Plants* (Travis et al. 1997), addressed the appropriateness of regulations (e.g., requirements for emergency planning and insurance) associated with spent fuel pool storage. The study also provided reasonable bounding estimates for offsite consequences for the most severe accidents, which would involve draining of the spent fuel pool (e.g., complete draining of the spent fuel pool occurs 12 days after shutdown of the reactor).

In 2001, the NRC issued NUREG–1738 (NRC 2001), *Technical Study of Spent Fuel Pool Accident Risk at Decommissioning Nuclear Power Plants*, which found that a postulated accident causing zirconium cladding fires could result in unacceptable offsite doses. Appendix F of this GEIS presents some results from NUREG–1738, including the largest number of early fatalities calculated (191). The large number of calculated fatalities was due, in part, to conservative assumptions for the ruthenium release (i.e., the release fraction is for a volatile fission product in an oxidic [rather than metallic] form), time of the accident (i.e., 30 days after shutdown of the reactor), and late evacuation of the public (i.e., evacuation is started after the release). More realistic assumptions (e.g., low ruthenium release, event occurs one year after shutdown), reduce the largest number of early fatalities from 191 to approximately two (NRC 2001). Although early fatalities are unacceptable, the annual likelihood for such an accident was estimated to be less than three chances in one million (NRC 2001). NUREG–1738 further states that “the risk at decommissioning plants is low and well within the Commission's safety goals. The risk is low because of the very low likelihood of a zirconium fire even though the consequences from a zirconium fire could be serious.” In arriving at this conclusion, NUREG–1738 considered a wide range of initiating events, including but not limited to, events that might lead to rapid loss of pool water (e.g., seismic events, cask drop, aircraft impact, and missiles generated by tornados). The low probability for these varied events to initiate a rapid loss of water from the pool is a direct result of the robustness of the structural design of the spent fuel pool.

As noted, spent fuel pools are massive structures constructed from thick, reinforced concrete walls and slabs designed to be seismically robust. Thus, the likelihood of major accidents at spent fuel pools resulting in offsite consequences is very remote. In particular, Appendix F determines that the environmental impacts from spent fuel pool fires are SMALL during the short-term storage timeframe based on the low risk of a spent fuel pool fire. The NRC is not aware of any additional studies that would cause it to question the low risk of spent fuel pool

accidents and thereby question the technical feasibility of continued safe storage of spent fuel in spent fuel pools for the short-term timeframe considered in the GEIS.

B.3.2 Technical Feasibility of Dry Cask Storage

The technical feasibility of dry cask storage is supported by years of experience and technical studies and NRC reviews that examined and confirmed the integrity of spent fuel and cladding under the controlled and benign environment within dry cask storage systems. The technical feasibility of these systems is further supported by the robustness of the structural design of the dry cask storage system against a variety of natural and human-induced challenges.

B.3.2.1 Low Degradation Rates of Spent Fuel in Dry Cask Storage

In the United States, spent fuel has been safely stored in dry casks for more than 25 years. In 1986, Virginia Power received a license for an at-reactor dry storage facility located at Surry Nuclear Power Plant. As of June 2014, there are operational ISFSIs at 64 sites in the United States. One operational ISFSI, at the GEH-Morris site, is a wet facility. The remaining ISFSIs are storing spent fuel in over 1,900 loaded dry casks. (see Section 2.1.2 in the GEIS for further details). As with wet storage, the overall experience with dry cask storage of similar fuel types, including the cladding, has been similar—slow degradation. In addition, spent fuel is cooled for a lengthy period in a spent fuel pool before being transferred into dry cask storage. NRC guidance regarding dry cask storage recommends a maximum cladding temperature of 400°C (752°F) and a dry, inert atmosphere to reduce the potential for significant degradation (NRC 2010c). Recent studies, including the following, have confirmed dry cask storage reliability:

1. A dry cask storage characterization project (Bare et al. 2001) examined and tested a dry cask storage system, the CASTOR V/21, and found "there was no evidence of cask, shielding, or fuel rod degradation during long-term (14 years) storage that would affect cask performance or fuel integrity." The project examined zirconium-clad fuel applicable for spent fuel with a burnup of 35 GWd/MTU. A subsequent study (Einziger et al. 2003), which examined spent fuel from the Bare et al. (2001) project, suggests that the spent fuel cladding could remain a viable barrier to fission product release during extended storage up to 100 years in a dry cask environment.
2. The IAEA status report *Understanding and Managing Ageing of Materials in Spent fuel Storage Facilities* (IAEA 2006) stated "[P]ower reactor fuel with zirconium alloy cladding has been placed into dry storage in approximately a dozen countries. The technical basis for satisfactory dry storage of fuel clad with zirconium alloys includes hot cell tests on single rods, whole assembly tests, demonstrations using casks loaded with irradiated fuel assemblies and theoretical analysis."

3. The Electric Power Research Institute (EPRI 1998) evaluated the data needs for long-term storage and reported that during normal storage of low-burnup spent fuel, “the lower radiation fields and estimated temperatures of 100–125°C after 20 years favor acceptable fuel behavior for extended storage.”

The NRC is aware that high-burnup and MOX fuel may be subject to increased degradation of the spent fuel and cladding that could cause further problems with handling, storing, and transporting spent fuel. With this increased usage, research has continued to improve understanding of degradation mechanisms affecting storage of spent fuel. Recent reports (e.g., NRC 2014; Hanson et al. 2012; IAEA 2011a; and Sindelar et al. 2011) have identified a variety of degradation mechanisms and discussed their potential effects on storage. For example, the mechanical integrity of the spent fuel cladding and assembly is important to ensure that handling and transportation of spent fuel can be conducted with relative ease. The mechanical designs of lower-burnup UOX and higher-burnup UOX or MOX fuel are very similar, but some of the after-irradiation properties of higher-burnup UOX and MOX are potentially significant in determining the rate of degradation or differences in performance. Differences in after-irradiation properties between lower-burnup UOX and higher-burnup UOX and MOX include higher fuel rod internal pressures and thinner cladding due to more cladding oxidation and hydride layer buildup causing higher cladding stress, higher decay heat, higher specific activity, and finer grain structure of the fuel pellet, potentially increasing the likelihood and consequences of an accident. Appendix I provides further discussion on the characteristics, storage, and transportation of high-burnup UOX and MOX spent fuel.

Although NRC regulations for dry cask storage allow for a licensing period of up to 40 years for both initial and renewed licenses, licensing periods approved for storage casks for high-burnup fuel have been limited to 20 years due to the more limited data available for high-burnup fuel. These storage times are sufficiently short and the degradation rates of spent fuel sufficiently slow that (1) significant storage, handling, and transportation issues are not expected to arise during a single license period and (2) should information collected during a license period identify any emerging issues and concerns, there would be sufficient time to develop regulatory solutions and incorporate them into future licensing periods.

Ongoing research into the extended storage of spent fuel is part of the NRC’s effort to continuously evaluate and update its safety regulations. As part of this effort, the NRC is examining the technical needs and potential changes to the regulatory framework that may be needed to continue licensing of spent fuel storage facilities over periods beyond 120 years. In 2014, the NRC published *Identification and Prioritization of the Technical Information Needs Affecting Potential Regulation of Extended Storage and Transportation of Spent Nuclear Fuel* (NRC 2014). This report considered high-burnup UOX fuel and MOX fuel. Further, international efforts are evaluating degradation mechanisms affecting handling, storage, and transportation of spent fuel (e.g., IAEA 2011a). The NRC, the DOE, other regulators, and the commercial power

industry have formed the Extended Storage Collaboration Program. The goal of this program is to better understand the degradation processes that could impact the storage of spent fuel. As new information becomes available, it will be considered in the development of canister design criteria and aging management requirements for the safe storage of spent fuel. Currently, EPRI is leading a multi-year research project, the majority of which is funded by DOE, to evaluate the safe storage of spent fuel in dry storage casks. EPRI will design and demonstrate dry cask technology at full scale for evaluating the condition of "high-burnup" spent fuel during storage. As research continues, if the NRC were to identify a concern with the safe storage of spent fuel, the NRC would evaluate the issue and take whatever action or make whatever change in its regulatory program necessary to protect public health and safety.

Based on available information and operational experience, degradation of the spent fuel should be minimal over the short-term storage timeframe if conditions inside the canister are appropriately maintained (i.e., consistent with the technical specifications for storage). Thus, the NRC expects that only routine maintenance will be needed over the short-term storage timeframe. Repackaging of spent fuel may be needed if storage continues beyond the short-term storage timeframe. In the GEIS, the NRC assumes that the dry casks would need to be replaced if storage continues beyond the short-term storage timeframe. Accidents associated with repackaging spent fuel are evaluated in Section 4.18 and the environmental impacts are SMALL because the accident consequences would not exceed the NRC accident dose standard contained in 10 CFR 72.106.

Spent fuel transfer operations can present challenges to operators and, in part, because of these challenges, transfer operations are conducted in enclosed, heavily shielded buildings with filters to reduce any potential releases. Although transfer operations at a current reactor would be conducted in the spent fuel pool and the dry transfer system would involve dry transfer, spent fuel transfer operations in either facility would occur within an enclosed, shielded building. Therefore, releases to the environment from handling operations within the spent fuel pool and the dry transfer system are expected to be similar. These operations routinely maintain public and occupational doses well within existing requirements. This is done despite variations in the facilities and equipment and the characteristics of the spent fuel being transferred. While these characteristics may vary, the safety regulations do not. In addition, the NRC requires that facilities and equipment be maintained to ensure safety functions are not compromised. Further, the NRC inspects operating facilities to verify compliance with requirements. As described in Section B.3.3.3 of this appendix, after the end of the reactor's licensed life for operation, the licensee would continue to store spent fuel onsite under either its 10 CFR Part 72 general license granted to 10 CFR Part 50 or Part 52 reactor licensees or a specific 10 CFR Part 72 license. During this time, the licensee would remain under the NRC's regulatory control and NRC inspections and oversight of storage facilities would continue. The NRC monitors the performance of ISFSIs (at decommissioned and shutdown reactor sites and at operating reactor sites) by conducting periodic inspections.

The opportunity to inspect spent fuel that has been placed into dry cask storage would occur during repackaging of the fuel. During the short-term timeframe, repackaging would occur, if needed, in the spent fuel pool, which would provide shielding and allow licensees to safely repackage the fuel. In the long-term and indefinite timeframes, repackaging would occur in the dry transfer system, which would be a shielded building. The NRC assumes replacement of dry casks after 100 years of service life; however, replacement times will depend on actual degradation observed during continued regulatory oversight for maintaining safety during continued storage. Studies and experience to date do not preclude a dry cask service life longer than 100 years. In addition, as described in Section 2.2.1.3 of the GEIS, in accordance with 10 CFR 72.42, ISFSI license renewal applications must include, among other things, (1) time-limited aging analyses that demonstrate that structures, systems, and components important to safety will continue to perform their intended safety function for the requested period of extended operation and (2) a description of the aging management program for management of issues associated with aging that could adversely affect structures, systems, and components important to safety. These requirements enhance confidence that spent fuel, including bare fuel, fuel in canisters, or damaged fuel that has been canned and stored in dry casks, could be retrieved for repackaging, if needed. Finally, regulatory experience shows that licensees have successfully dealt with damaged fuel. In the most extreme example, the damaged fuel from the core of Three Mile Island, Unit 2 (TMI-2), was removed and safely placed into storage. If this type of fuel can be successfully moved and managed, then it is reasonable to assume that damaged spent fuel in casks can be handled, if necessary. Although a commercial dry transfer system is currently not operating in the United States, construction and operation of a dry transfer system, including the handling of damaged fuel, can be accomplished with current technology (further information provided in Section 2.2.2.1 – Construction and Operation of a Dry Transfer System).

B.3.2.2 Robust Design of Dry Cask Storage Systems

Dry cask storage systems are passive systems (i.e., relying on natural air circulation for cooling) that are inherently robust, massive, and highly resistant to damage. To date, the NRC and licensee experience with ISFSIs and cask certification indicates that spent fuel can be safely and effectively stored using dry cask storage technology. There have not been any safety issues with dry cask storage.

In addition, the NRC's technical review supporting issuance of Materials License No. SNM–2513 for the Private Fuel Storage, LLC (PFS) facility has confirmed the technical feasibility of continuing storage at an away-from-reactor ISFSI under 10 CFR Part 72 (NRC 2006a). While issues extraneous to safety and protection of the environment have, to date,

prevented the licensee from going forward with the project,[5] the NRC's extensive review of safety and environmental issues associated with construction and operation of the PFS facility provides further information supporting the technical feasibility of safe spent fuel storage at an away-from-reactor ISFSI for long periods following storage at a reactor site (i.e., in a spent fuel pool or at-reactor ISFSI).

The NRC has renewed three specific ISFSI licenses for an extended 40-year period. Because at that time Part 72 only provided for a renewal period of 20 years, an exemption was granted as part of the NRC's review of the safety of renewing Part 72 license for 40 years. The NRC published a final rule on February 16, 2011, to clarify the processes for the renewal of ISFSIs operated under the general license provisions of 10 CFR Part 72, for renewal of the Certificate of Compliance for dry cask storage systems, and for extending the license and renewal terms to 40 years (76 FR 8872). In these cases, the NRC's technical review has encompassed the applicant's evaluation of aging effects on the structures, systems, and components important to safety, supplemented by the applicant's aging management program. These comprehensive reviews support the technical feasibility of continued safe storage of spent fuel in these ISFSIs and thus reaffirm the technical feasibility of safe, interim dry storage for an extended period. While these license renewal cases address storage at an ISFSI for a period of up to 80 years (i.e., up to 40-year initial license, plus 40-year renewal), studies performed to date (e.g., Einziger et al. 2003; EPRI 2002; 55 FR 38472) have not identified any issues that would call into question the technical feasibility of long-term use of dry storage for low-burnup spent fuel.

In 2007, the NRC published a pilot probabilistic risk assessment methodology (NRC 2007) that identified the dominant contributors to risk associated with a welded-canister dry-spent-fuel-storage system at a specific boiling water reactor site. The NRC study developed and assessed a comprehensive list of initiating events, including dropping the cask during handling and external events during onsite storage (e.g., earthquakes, floods, high winds, lightning strikes, accidental aircraft crashes, and pipeline explosions) and reported that the analysis indicates that the overall risk of dry cask storage was found to be extremely low. (The NRC determined that the estimated aggregate risk is an individual probability of a latent cancer fatality of 1.8×10^{-12} during the period encompassing the initial cask loading and first year of service and 3.2×10^{-14} per year during subsequent years of storage [NRC 2007]).

Several characteristics of dry cask storage contribute to the low risk determined by the NRC study. First, these systems are passive. Second, they rely on natural air circulation for cooling.

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[5] Although a license was issued, the PFSF has not yet been constructed. However, the NRC determined, based on its review of the application, that there is reasonable assurance that if the PFSF is constructed (1) the activities authorized by the license can be conducted without endangering the health and safety of the public and (2) these activities will be conducted in compliance with the applicable regulations of 10 CFR Part 72 (NRC 2006a).

Third, their inherently robust, massive concrete and steel structure is highly resistant to damage. The robustness of these dry cask storage systems has been tested by significant challenges (e.g., the August 23, 2011 Mineral, Virginia earthquake that affected the North Anna Nuclear power plant and the March 11, 2011 earthquake and subsequent tsunami that damaged the Fukushima Dai-ichi nuclear power plant). Neither event resulted in significant damage to the dry cask storage containers or the release of radionuclides (VEPCO 2011; INPO 2011).[7]

Thus, technical studies and practical operating experience to date confirm the physical integrity of dry cask storage structures and thereby demonstrate the technical feasibility of continued safe storage of spent fuel in dry cask storage systems for the time periods considered in the GEIS. Further, the NRC expects that only routine maintenance will be needed over the short-term storage timeframe. Repackaging of spent fuel may be needed if storage continues beyond the short-term storage timeframe. The NRC is not aware of any issue that would cause it to question the technical feasibility of continued safe storage of spent fuel in dry casks for the timeframes considered in the GEIS. Further, the NRC continues to evaluate aging management programs and to monitor dry cask storage so that it can update its service life assumptions as necessary and consider any circumstances that might require repackaging of spent fuel earlier than anticipated.

B.3.3 Regulatory Oversight of Wet and Dry Spent Fuel Storage

A strong regulatory framework that includes both regulatory oversight and licensee compliance is important to the continued safe storage of spent fuel. As part of its oversight, the NRC can issue orders and new or amended regulations to address emerging issues that could impact the safe storage of spent fuel. This section provides a discussion of how the NRC's regulatory program has addressed potential safety and security concerns and routine operations. The environmental impact analysis in the GEIS relies upon the current regulatory framework, which includes whatever license amendments, orders, and rulemaking becomes necessary to protect public health and safety. These ongoing improvements to the NRC's regulatory structure are reflected in the NRC's upgrade of safety, environmental, and security requirements following historic events, (e.g., the regulatory changes following the TMI-2 accident in 1979; safety and security upgrades following the September 11, 2001 terrorist attacks; and the Task Force recommendations and improvements to safety following the March 11, 2011 earthquake and subsequent tsunami that crippled the Fukushima Dai-ichi nuclear power plant). These regulatory changes demonstrate the NRC's capability for prompt and vigorous response to new developments that warrant increased regulatory attention. Thus, the vitality and evolution of the NRC's regulatory requirements support a reasonable conclusion that continued storage, even

[7] Dry casks at the Fukushima Dai-ichi nuclear power plant are stored in a shared dry cask storage building.

over extended periods of time beyond those regarded as most likely, will continue to be safe with the same or fewer environmental impacts.

B.3.3.1 Regulatory Actions for Routine Operations, Accidents, and Terrorist Activity

As part of its oversight, the NRC can issue orders and new or amended regulations to address emerging issues that could impact the safe storage of spent fuel. An example of the NRC's regulatory oversight is the NRC's actions following the TMI-2 accident in 1979. First, the NRC created a Bulletin and Orders Task Force to assure the immediate safety of all other operating power reactors. Next, the NRC established the TMI-2 Lessons Learned Task Force to identify and evaluate safety concerns requiring prompt licensing actions for operating reactors, beyond the immediate actions announced by the earlier Task Force. A set of short-term recommendations was published as NUREG–0578 in July 1979 (NRC 1979). The NRC then assessed recommendations that "would provide a comprehensive and integrated plan for all actions necessary to correct or improve the regulation and operation of nuclear facilities." This "TMI-2 Action Plan" was published as NUREG–0660 in May 1980 (NRC 1980a). These action items led NRC to issue a list of "Requirements for New Operating Licenses," published in NUREG–0694 (NRC 1980b), which was later clarified and superseded by NUREG–0737 (NRC 1980c). Finally, after issuance of TMI-2 Action Plan requirements, the NRC codified new reactor requirements by regulation (46 FR 26491).

Another example, following the terrorist attacks of September 11, 2001, the NRC undertook an extensive reexamination of spent fuel safety and security issues. In 2002, the NRC issued orders to licensees that required power reactors in decommissioning, spent fuel pools, and ISFSIs to enhance security and improve their capabilities to respond to, and mitigate the consequences of, a terrorist attack. For example, these orders required additional security measures, including increased patrols, augmented security forces and capabilities, and more restrictive site-access controls to reduce the likelihood of a successful terrorist attack. In 2007, the NRC issued a final rule revising the Design Basis Threat,[8] which also increased the security requirements for power reactors and their spent fuel pools (72 FR 12705). More recently, in 2009, the NRC issued a final rule to further improve security measures at nuclear power reactors, including at spent fuel pools (74 FR 13926). This rule included improvements to security measures, such as enhancements to cyber security plans, facilitation of consistent application of preparatory actions with respect to air attacks, integration of the access authorization and security program requirements, and additional requirements for unarmed security personnel to ensure these personnel meet the minimum physical requirements commensurate with their duties.

[8] A design basis threat provides a general description of the attributes of potential adversaries who might attempt to commit radiological sabotage or theft or diversion against which licensee's physical protection systems must defend with high assurance.

Section 4.19 of the GEIS describes the environmental impacts of potential acts of sabotage or terrorism involving the continued storage of spent fuel. This section acknowledges that as the immediate hazard posed by the high radiation levels of spent fuel diminishes over time, depending on burnup, so does the deterrent to handling by unauthorized persons. The NRC will consider this type of information in evaluating whether additional security requirements are warranted in the future.

The most recent examples of the NRC's response to unexpected developments are the additional requirements that the NRC has already imposed or is considering in response to the March 11, 2011 earthquake and subsequent tsunami that resulted in extensive damage to the six-unit Fukushima Dai-ichi nuclear power plant in Japan. On March 12, 2012, the NRC issued multiple orders and a request for information to all of its nuclear power plant licensees (NRC 2012a). The request for information was issued to all licensees to determine whether nuclear plant licenses should be modified, suspended, or revoked. The purpose of the request for information was to re-evaluate seismic and flooding hazards at operating reactor sites and to determine whether appropriate staffing and communication can be relied upon to coordinate event response during a prolonged station blackout event, as was experienced at Fukushima Dai-ichi. Section 4.18 and Appendix F provide further details regarding the NRC's orders and requests for information in response to the Fukushima event.

Another aspect of the NRC's regulatory program for continued storage at reactors and other licensed facilities involves generic communications. Generic communications include, but are not limited to, generic letters, bulletins, information notices, safeguards advisories, and regulatory issue summaries. Generic letters request licensee actions or information to address issues regarding emergent or routine matters of safety, security, safeguards, or environmental significance. Bulletins request licensee actions or information to address significant issues regarding matters of safety, security, safeguards, or environmental significance that have great urgency. Information notices are used to communicate operating or analytical experience to the nuclear industry. The industry is expected to review the information for applicability and consider appropriate actions to avoid similar problems. Regulatory issue summaries are used to communicate and clarify the NRC's technical or policy positions on regulatory matters.

For example, Information Notice 2012–20 (NRC 2012b) informed licensees about the potential for chloride-induced stress corrosion cracking of austenitic stainless steel and maintenance of dry cask storage system canisters. Although an immediate safety concern did not exist, the NRC alerted its licensees and certificate holders that their monitoring programs need to address this concern as part of an aging management program so that appropriate actions (e.g., maintenance) would be taken before there were any impacts.

Another example is Information Notice 2009-26, *Degradation of Neutron-Absorbing Materials in the Spent Fuel Pool*, with respect to criticality safety for pool storage (NRC 2009a). NRC licensees use various methods to meet subcriticality requirements in the spent fuel pool

specified by 10 CFR 50.68 or 10 CFR Part 50 Appendix A, General Design Criterion 62. Most spent fuel pools now store spent fuel assemblies in high-density racks, which incorporate neutron absorber materials into the rack walls. These neutron absorber materials, especially boraflex, can degrade enough to lose their neutron-absorbing capabilities and challenge subcriticality requirements (requirements to prevent an uncontrolled chain reaction). Due to this degradation, many licensees now employ other means to meet subcriticality requirements (e.g., spent fuel loading patterns, fuel burnup credit, control rods or other neutron poisons contained within spent fuel bundles, soluble boron in the pool water, or some combination of these methods). The NRC issued Information Notice 2009-26 to all operating reactors licensees and construction permit holders in October 2009 (NRC 2009a). The NRC continues to monitor how licensees are addressing the degradation issue. Most recently, on March 11, 2014, the NRC issued a draft generic letter for public comment that, if finalized, would request information from licensees to allow the NRC to "determine if the degradation of the neutron-absorbing materials in the SFP is being managed to maintain reasonable assurance that the materials are capable of performing their intended safety function, and if the licensees are incompliance with the regulations" (79 FR 13685).

B.3.3.2 Regulatory Oversight of Spent Fuel Pool Leaks

Spent fuel pool design and operational control requirements in NRC regulations make it unlikely that a leak will remain undetected long enough to result in public health and safety or environmental concerns. Long-standing design requirements include but are not limited to general design criteria in 10 CFR Part 50, Appendix A that focus on fuel storage and handling and radioactivity control (e.g., General Design Criterion 61). Operational controls include requirements for control of effluents and release of radioactive materials such as dose limits found in 10 CFR 20.1301 and design objectives found in 10 CFR Part 50, Appendix I.

There are also requirements that are new or have been updated in response to recent operational experience and related studies by NRC task forces. For example, a 2006 report by NRC's Liquid Radioactive Release Lessons Learned Task Force made 26 specific recommendations for improvements to NRC regulatory programs (NRC 2006b). In 2010, the NRC Groundwater Task Force reevaluated the recommendations of the 2006 Task Force (NRC 2010d). A review of the Groundwater Task Force recommendations by NRC senior management concluded that further action was warranted (NRC 2011d). These studies have influenced specific changes to NRC requirements and guidance. For example:

- In June 2008, the NRC issued Regulatory Guide 4.21, *Minimization of Contamination and Radioactive Waste Generation: Life-Cycle Planning* (NRC 2008). The purpose of this regulatory guide is to present guidance that will assist applicants covered by 10 CFR 20.1406, "Minimization of contamination," in effectively implementing this licensing requirement.

- In June 2009, the NRC issued revision 2 to Regulatory Guide 4.1 (NRC 2009b) provides guidance to licensees for detecting, evaluating, and monitoring releases from operating facilities via unmonitored pathways; to ensure consistency with current industry standards and commercially available radiation detection methodology; to clarify when a licensee's radiological effluent and environmental monitoring programs should be expanded based on data or environmental conditions; and to ensure that leaks and spills are detected before radionuclides migrate offsite via an unmonitored pathway.
- In July 2011, the NRC promulgated its Decommissioning Planning Rule, which added 10 CFR 20.1406(c) and modified 10 CFR 20.1501(a) and (b) (76 FR 35512). This rule requires all licensees to establish operational practices to minimize site contamination and perform reasonable subsurface radiological surveys and sets forth new financial assurance requirements.
- In December 2012, the NRC published Regulatory Guide 4.22, *Decommissioning Planning During Operations*, which provides methods acceptable to the NRC to use in implementing portions of the Decommissioning Planning Rule (NRC 2012c).

Appendix E of the GEIS provides a detailed description and evaluation of the historical data on spent fuel pool leaks, discusses ongoing and future monitoring activities and corrective actions, and analyzes potential environmental impacts that may occur during the short-term timeframe during which spent fuel storage in pools will continue. Appendix E concludes that the potential environmental impacts from spent fuel pool leakage would be SMALL.

B.3.3.3 Dry Cask Storage

Consistent with the NRC's regulatory framework for continued safe spent fuel storage in dry casks, reactor and ISFSI licensees have acted prudently to safely manage their spent fuel. In the late 1970s and early 1980s, the need for alternative storage began to grow as spent fuel pools at many nuclear reactors began to reach their licensed capacity. License amendments to allow spent fuel pool re-racking, fuel-pin consolidation, and specific or general licenses for onsite dry cask storage have been successfully employed to increase onsite storage capacity. As discussed previously, there are currently operational ISFSIs at 64 sites. The NRC is successfully regulating seven fully decommissioned reactor sites that contain ISFSIs licensed under either the general or specific license provisions of 10 CFR Part 72.[11]

After the end of a reactor's licensed life for operation, the licensee would continue to store spent fuel onsite under either the 10 CFR Part 72 general license granted to 10 CFR Part 50 and Part 52 reactor licensees or a specific 10 CFR Part 72 license. During this time, the licensee

[11] These reactor sites include Maine Yankee, Yankee Rowe, Connecticut Yankee (also known as Haddam Neck), Fort St. Vrain, Rancho Seco, Trojan, and Big Rock Point.

would remain under the NRC's regulatory control and NRC inspections and oversight of storage facilities would continue. The NRC monitors the performance of ISFSIs (at both decommissioned and shutdown reactor sites and operating reactor sites) by conducting periodic inspections. When conducting inspections at these ISFSIs, NRC inspectors follow the guidance in NRC Inspection Manual Chapter 2690, *Inspection Program for Dry Storage of Spent Reactor Fuel at Independent Spent Fuel Storage Installations and for 10 CFR Part 71 Transportation Packages* (NRC 2012d).

The current regulatory framework for storage of spent fuel allows for multiple license renewals, subject to aging management analysis and planning. In early 2011, the Commission published a final rule that amended 10 CFR Part 72 to increase the initial and renewal terms for specific ISFSI licenses from "not to exceed 20 years" to "not to exceed 40 years" (76 FR 8872). The Commission concluded that, with appropriate aging management and maintenance programs, license terms not to exceed 40 years are reasonable and adequately protect public health and safety. An applicant for a storage license renewal must provide appropriate technical bases for identifying and addressing aging-related effects and must develop specific aging management plans to justify extended operations of ISFSIs. The regulatory framework for storage is supported by well-developed regulatory guidance; voluntary domestic and international consensus standards; research and analytical studies; and processes for implementing licensing reviews, inspection programs, and enforcement oversight.

B.3.3.4 Summary of Information on Regulatory Oversight

The NRC will continue its regulatory control and oversight of spent fuel storage at both operating and decommissioned reactor sites under both specific and general 10 CFR Part 72 licenses. Decades of operating experience and ongoing NRC inspections demonstrate that these reactor and ISFSI licensees continue to meet their obligation to safely store spent fuel in accordance with the NRC's requirements. If the NRC were to find noncompliance with these requirements or otherwise identify a concern with the safe storage of the spent fuel, the NRC would evaluate the issue and take necessary action or change its regulatory program to protect the public health and safety and the environment.

As noted in the preceding paragraphs, licensees have continued to develop and successfully use onsite spent fuel storage capacity in the form of spent fuel pool and dry cask storage in a safe and environmentally sound fashion. Based on the preceding discussion, the NRC believes that for the storage timeframes considered in the GEIS, regulatory oversight will continue in a manner consistent with NRC's regulatory actions and oversight in place today to provide for continued safe storage of spent fuel as long as spent fuel needs to be stored.

B.3.4 Continued Institutional Controls

As discussed in the previous sections of this appendix, continued safe storage of spent fuel requires both the technical feasibility of safe storage and a regulatory framework that provides for monitoring and oversight to address the potential for evolving issues. To ensure adequate protection of public health and safety, the institutional controls provided by the NRC's regulatory structure and that of sister agencies, as well as by Federal, State and local governments in general, must be maintained over time. The GEIS takes the following approach to institutional controls:

1. the GEIS's evaluation of reasonably foreseeable environmental impacts of continued storage requires an assumption that institutional controls will be maintained;
2. the most reasonably foreseeable assumption is that institutional controls will continue;
3. accidents provide a helpful surrogate for analysis of a temporary lapse in institutional controls, including perspectives on the environmental implications of such a lapse; and
4. although too remote to calculate meaningfully, a permanent loss of institutional controls would likely have catastrophic consequences.

A detailed discussion for each of these topics is provided below.

1. An evaluation of reasonably foreseeable environmental impacts in the GEIS requires an assumption that institutional controls will be maintained

In *New York v. NRC*, the Court of Appeals held that because the NRC had not demonstrated that the unavailability of a repository was "remote and speculative," the National Environmental Policy Act (NEPA) required the NRC to analyze the environmental impacts of continued storage in the absence of a repository. The NRC believes that, if geologic disposal were not possible, national spent fuel policy would change but would not default to relying on the storage facilities as they currently exist—the design of facilities and the regulations governing those facilities would change to accommodate the new policy. Further, the NRC is not in a position to predict how the policy would change or what technical advancements would become available to serve a new national policy if geologic disposal were not feasible or achievable by consensus. Analyzing the consequences of failing to secure a repository requires assumptions about what indefinite continued storage would encompass. Because the current methods of continued storage employ institutional controls, the NRC considered whether it was reasonable to assume that institutional controls would remain in place in the timeframes being considered, and, as explained below, concluded that the assumption is reasonable for the purposes of this GEIS. While the NRC does not believe that the indefinite storage scenario described in the GEIS is likely, the NRC has analyzed this scenario in the GEIS to provide a conservative picture of the environmental impacts should a repository not become available by the end of the long-term timeframe.

As stated in Chapter 1 of this GEIS, the Federal government, by national policy set forth in the Nuclear Waste Policy Act, has assumed responsibility for the permanent disposal of high-level radioactive waste and spent fuel. The Nuclear Waste Policy Act specifies that the cost of both interim storage and permanent disposal is the responsibility of the generators and owners of the waste. Further, the Nuclear Waste Policy Act defines the current national strategy for disposition of spent fuel as disposal in a geologic repository; the geologic repository strategy was recently reaffirmed by the Blue Ribbon Commission on America's Nuclear Future (BRC 2012).

In response to the Blue Ribbon Commission's report (BRC 2012), the DOE expressed its intent to provide a repository by 2048 (DOE 2013), which is about 10 years before the end of the short-term timeframe for the oldest spent fuel storage facility within the scope of this analysis. In this GEIS, the NRC concludes that a repository is most likely to be available by the end of the short-term timeframe, and failing that, likely to be available by the end of the long-term timeframe. In the event a repository could not be sited by the end of the long-term timeframe, the NRC has concluded that it is not reasonable to assume that national policy would default to complete inaction so as to leave spent fuel in dry casks unprotected, much less unattended or ultimately forgotten. However, because an alternate path forward is unknown at this point, the NRC has not attempted to forecast a different solution and assumes that continued storage continues indefinitely.

Should the national policy change from geologic disposal to permanent storage (i.e., onsite or away-from-reactor "disposal" in facilities that resemble ISFSIs), the NRC expects that planning and decision-making for permanent storage of spent fuel would take into account the appropriate balance of engineering design and institutional controls to address the challenges presented by permanent storage. There is no national historic precedent and, more particularly, no regulatory history of nuclear materials to suggest that the Federal government, including the NRC in its assigned role under the Atomic Energy Act, would not engage in planning and decision-making regarding whatever further changes or enhancements would be necessary to accommodate permanent storage, in the unlikely event that option was adopted. Should national policy change to a policy of permanent storage, the NRC believes that significant regulatory changes and design modifications would be required to transfer spent fuel to offsite facilities or convert onsite continued storage facilities to onsite permanent storage facilities. Further, even if a repository does not become available, the NRC believes that, based on the factors discussed in the next section, institutional controls will be maintained as long as the spent fuel needs to be stored.

With respect to costs, the NRC acknowledges that, because of delays in the siting and licensing of a repository, the Federal government bears an increasing share of the financial responsibility for storage costs. Although the annual costs for continued storage are manageable, cumulative costs will continue to increase. The Federal government has estimated it will pay a total of approximately $20 billion in damage awards and settlements by the year 2020 and $500 million

per year after that if DOE does not accept fuel by 2021 and spent fuel continues to accumulate at reactor sites (GAO 2013). Thus, the escalating costs of continued storage provide incentive for the Federal government to implement the national policy for disposal of spent fuel in a deep geologic repository.

The assumption that institutional controls will continue enables an appropriate and reasonable evaluation of the environmental impacts of continued storage over an indefinite timeframe. Absent the stability and predictability that follows from institutional controls, including but not limited to NRC licensing and regulatory controls, few impacts could be reliably forecast. The "hard look" required by NEPA would quickly become unfocused, highly speculative, and ill-defined. Analyzing the impacts that might result from a permanent and total loss of institutional controls would require NRC to reach unsupportable conclusions about how and when our nation, and its government, institutions, and social cohesiveness might degrade or even collapse. Such speculation would preclude meaningful calculations of impacts for the timeframes envisioned in the GEIS.

2. The assumption that institutional controls continue is reasonable

Consistent with NEPA's rule of reason, which provides that agencies conduct an analysis according to the usefulness of the information to the decision-maker and full disclosure to the public of predictable benefits and impacts, this GEIS assumes that institutional controls at any storage site are maintained. This assumption is reasonable for two reasons: First, in any timeframe it would be illogical for any government at any level to abandon the storage facilities, given the particular hazards of the fuel. Continued storage is designed to allow the eventual transport of the spent fuel to a repository, not to permanently sequester the material from the environment without continued active oversight and maintenance. Second, these highly visible storage facilities are much less likely than buried geologic repositories to simply be forgotten.

Spent fuel is highly hazardous, requiring robust containment structures to minimize exposure risks. Spent fuel in storage facilities on the surface of the earth presents a visible hazard that requires active oversight to ensure safety and security measures are maintained and functioning as designed. Storage facilities remain under license and have aging management programs to support their continued maintenance and monitoring. Thus, the visibility of storage facilities and the hazards of spent fuel strongly support the reasonableness of assuming the continuation of institutional controls throughout all of the timeframes analyzed in the GEIS. While changes may occur over time to governments or society, highly visible, hazardous facilities are unlikely to be left abandoned or forgotten. As a result, it is a reasonable assumption that any government would, in the interest of its citizenry, ensure that appropriate oversight (e.g., monitoring, maintenance, and replacement of facilities as needed) remains in place, consistent with radiation protection principles and regulatory restrictions, until final disposition of the spent fuel occurs. Accordingly, the NRC has determined that the assumption of continued institutional controls is reasonable in each of the timeframes considered in the GEIS.

In contrast, consideration of the loss of institutional controls in the context of disposal of spent fuel—as in DOE's Yucca Mountain environmental impact statement (EIS), for example—is not directly applicable to storage: NRC regulations for deep geologic disposal of spent fuel recognize there is a point when the repository ceases operation, is permanently closed, and the license terminated. After permanent closure, regulations specify institutional controls (e.g., the requirements to place markers to identify what is buried deep below the surface of the earth and to maintain records regarding the hazard). However, these institutional controls are part of a defense-in-depth approach to disposal; the facility design is not permitted to rely on those institutional controls to meet postclosure safety requirements.

Additionally, as identified in the public comments for this proceeding (see Appendix D of the GEIS), a repository applicant is required to prepare a stylized calculation to evaluate the consequences should humans inadvertently disrupt the repository (see 10 CFR 63.322). These requirements for disposal address the situation where human activities could occur at a disposal site that is no longer recognizable at the earth's surface following waste burial, permanent closure of the facility, and license termination. However, in contrast to underground disposal facilities, aboveground storage installations are not designed to be abandoned and will remain highly visible on the earth's surface. As explained previously, the visibility and purpose of temporary storage facilities differ significantly from those of permanent disposal facilities, supporting the reasonableness of assuming that institutional controls over continued storage facilities will be maintained.

The NRC recognizes information presented by the National Academies National Research Council and others regarding the durability of institutional controls (e.g., NAS 1995, 2000). The NRC is also aware of international reports that discuss the durability of institutional controls (e.g., NEA 2006, IAEA 2011b). However, this commentary does not conclude that a permanent loss of institutional controls is likely or that effective government and governmental oversight of continued storage will cease in the distant future. Rather, these documents focus on developing plans and strategies regarding what should be done today to address future uncertainty due, in part, to institutional controls.

For example, the Board on Radioactive Waste Management, in its study on long-term institutional management, stated: "No plan developed today is likely to remain protective for the duration of the hazards. Instead, long-term institutional management requires periodic, comprehensive reevaluation of those legacy waste sites still presenting risk to the public and the environment to ensure that they do not fall into neglect and that advantage is taken of new opportunities for their further remediation" (NAS 2000). While regulations may need to be updated over time, the NRC does not view possible future regulatory updates as an impediment to a current understanding of likely environmental impacts of continued storage. Further, future regulatory development would be expected to be undertaken to enhance and improve the effectiveness of regulatory oversight.

3. Accident analysis provides a perspective on the environmental impacts of a temporary lapse of institutional controls

The GEIS considers the environmental impacts of accidents during continued storage (e.g., certain cask drop events) in Section 4.18. These accidents, for the purposes of this NEPA analysis may serve as a surrogate or proxy for the temporary loss of institutional controls, and the impacts of these accidents are representative of impacts from a temporary loss of institutional controls. An accident condition approximates a limited period during which institutional controls are less than effective, after which the NRC expects that institutional controls and oversight would resume. Consequences from accidents resulting in small releases represent a lapse in more routine maintenance tasks, whereas accidents resulting in significant radioactive releases constitute a reasonable surrogate to evaluate consequences that might result from hypothetical acts of radiological sabotage or terrorism in the indefinite timeframe. Consideration of accident consequences thereby provides a reasonable basis for understanding the consequences of continued storage should institutional controls prove temporarily ineffective.

Given the physical characteristics of spent fuel, in most cases, the level of institutional controls necessary for safety would diminish over time and the consequences associated with accidents made possible by lapses in institutional controls would be expected to decrease with the passage of time. The thermal output of spent fuel decreases by approximately a factor of ten in the first 100 years after it is removed from the reactor, which means that maintenance activities and related institutional controls could be adjusted, as appropriate, to account for lower thermal loads. Therefore, the consequences of ineffective institutional controls will diminish over time because lower thermal loads should reduce the need for maintenance activities to maintain safety and lower radioactivity should reduce the consequences of releases of spent fuel. In contrast, institutional controls with respect to security many not diminish. As discussed in Section 4.19.2 of the GEIS, because spent fuel radiation levels will decrease over time, spent fuel could become more susceptible to theft or diversion (i.e., a more attractive target to individuals with malevolent intent). For this reason, additional security requirements may be necessary in the future if spent fuel remains in storage, to ensure that risk posed due to theft or diversion remains very low.

4. Impacts of loss of institutional controls

Some comments recommended that the NRC consider the evaluation of the loss of institutional controls based, in part, on DOE's Yucca Mountain EIS (DOE 2008b), which included an analysis for the loss of institutional controls for storage facilities under the no-action alternative. The NRC notes that DOE's proposed action in that instance was the construction of a repository and that, as a result, analysis of the no-action alternative was required by NEPA. Permanent disposal of spent fuel is a DOE responsibility, and DOE's analysis was designed to evaluate the environmental impacts of not meeting that responsibility. DOE evaluated the storage of the total

volume of high-level waste (i.e., 70,000 MTU) that would be disposed at the repository and, as a means of evaluating what would happen if it took no action, it considered the consequences of a simultaneous loss of institutional controls at 72 commercial and 5 DOE storage sites. In contrast, this GEIS considers the environmental impacts of continued storage at a single generically profiled commercial facility. While the DOE analysis may have sufficed for DOE's Yucca Mountain EIS, the NRC does not believe that the passive scenario assumed as part of the no-action alternative there provides a meaningful method of analyzing the consequences of indefinite storage for purposes of analyzing continued storage in this GEIS.

DOE's analysis evaluates degradation of the storage structures in the absence of human intervention (i.e., that neither government nor local residents, or even malevolent forces, would respond to the degradation in any fashion over a 10,000-year period). DOE did not state that its analysis of the loss of institutional controls represents the reasonably foreseeable impacts of permanent aboveground storage. To the contrary, DOE stated that neither of the no-action scenarios is likely to occur (DOE 2002). However, DOE's Yucca Mountain EIS (DOE 2008b) concluded that the consequences of the potential loss of institutional controls could be "catastrophic" in some resource areas.

As discussed previously, merely assuming loss of institutional controls in the distant, but undefined, future is not enough for the NRC to reasonably foresee when and how the loss of institutional controls might occur, and the consequences of that loss, with the kind of detailed and scientifically supportable analysis of resource impacts that the GEIS provides in every other respect for decision-makers and the public. Rather, the NRC would need to hypothesize the extent to which controls must fail before spent fuel would be effectively abandoned. The difficulty in predicting future consequences is further compounded by the lack of any credible way to foresee the combination of human and natural forces that might act on abandoned storage casks and cause a release. In addition, the baseline human environment becomes increasingly unpredictable the further out in time projections are made.

Nevertheless, the NRC can state broadly that, if institutional controls should be lost through a gradual dissolution of government or an apocalyptic event, unmitigated physical deterioration of spent fuel casks and cladding over decades, if not centuries, would eventually expose radionuclides to the environment. While the consequences—as explained above—are unpredictable, the NRC can state qualitatively that the consequences of such a catastrophe to the environment and public health could be similar to the impacts DOE analyzed for the no-action alternative (scenario 2—permanent loss of institutional controls) in its Yucca Mountain EIS (assuming a similar number of facilities were considered). Thus, in the event of a permanent loss of institutional controls, the resulting consequences to the environment across nearly all resource areas would be clearly noticeable and destabilizing.

B.3.5 Summary of Technical Feasibility of Continued Storage

As discussed previously, the NRC believes that it is reasonable to assume that the storage of spent fuel in any combination of storage in spent fuel pools or dry casks will continue as a licensed activity under regulatory controls and oversight. Licensees have continued to develop and successfully use onsite spent fuel storage capacity in the form of spent fuel pool and dry cask storage in a safe and environmentally sound fashion. Technical understanding and operational experience continues to support the technical feasibility of safe storage of spent fuel in spent fuel pools and in dry casks over long periods of time (e.g., slow degradation of spent fuel during storage in spent fuel pools and dry casks; engineered features of storage pools and dry casks to safely withstand accidents caused by either natural or man-made phenomena). In addition, regulatory oversight has been shown to enhance safety designs and operations as concerns and information evolve over time (e.g., safety enhancements made after the Three Mile Island accident in 1979, the September 11, 2001 terrorist attacks, and the March 11, 2011 Fukushima Dai-ichi disaster.

Based on the technical information and the national and international experience with wet and dry storage of spent fuel, the NRC concludes it is technically feasible to safely store spent fuel in either wet or dry storage for the short-term storage timeframe with only routine maintenance (i.e., no large-scale replacement of spent fuel pools or dry cask storage systems).

In the GEIS, the NRC assumes that after the short-term storage timeframe, spent fuel is stored in dry casks. Further, as discussed previously, the NRC concludes that there is no technical reason that spent fuel cannot be safely stored in dry casks beyond the short-term storage timeframe. As discussed in this appendix, the degradation rates of spent fuel are low under dry storage conditions and the probability of accidents with large consequences are very low. Storage of spent fuel beyond the short-term storage timeframe would continue under an approved aging management program to ensure that monitoring and maintenance are adequately performed. Repackaging of spent fuel may be needed if storage continues beyond the short-term storage timeframe. In the GEIS, the NRC assumes the replacement of dry casks after 100 years of service life; however, actual replacement times will depend on actual degradation observed during continued regulatory oversight for maintaining safety during continued storage. Studies and experience to date do not preclude a dry cask service life longer than 100 years. Accidents associated with repackaging spent fuel are evaluated in Section 4.18 and the environmental impacts are SMALL because the accident consequences would not exceed the NRC accident dose standard contained in 10 CFR 72.106. The NRC concludes it is technically feasible to continue to store spent fuel beyond the short-term storage timeframe, which may include activities to repackage spent fuel.

Section 4.19 of the GEIS describes the environmental impacts of potential acts of sabotage or terrorism involving the continued storage of spent fuel. This section acknowledges that as the immediate hazard posed by the high radiation levels of spent fuel diminishes over time so does

the deterrent to handling by unauthorized persons. The Blue Ribbon Commission's report noted that "over long time periods (perhaps a century or more, depending on burnup and the level of radiation that is deemed to provide adequate self-protection), the fuel could become more susceptible to possible theft or diversion (although other safeguards would remain in place). This in turn could require changes to the security requirements for older spent fuel. Extending storage to timeframes of more than a century could thus require increasingly demanding and expensive security protections at storage sites." If necessary, the NRC will issue orders or enhance its regulatory requirements for ISFSI security, as appropriate, to provide adequate protection of public health and safety and the common defense and security.

B.4 Conclusions

This appendix evaluates the technical feasibility of continued storage and repository availability, including national and international experience with storage and disposal of spent fuel. Based on the information and experience presented in this appendix, the NRC concludes that (1) a geologic repository is technically feasible; (2) the time period needed to develop a repository is approximately 25 to 35 years; (3) continued safe storage of spent fuel in spent fuel pools for the short-term timeframe is technically feasible; and (4) continued safe storage of spent fuel in dry casks for the timeframes considered in the GEIS is technically feasible. Further, the NRC concludes that a strong regulatory framework including both regulatory oversight and licensee compliance is important to the continued safe storage of spent fuel. As discussed in this appendix, the regulatory framework for storage is supported by well-developed regulatory guidance; voluntary domestic and international consensus standards; research and analytical studies; and processes for implementing licensing reviews, inspection programs, and enforcement oversight.

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NUREG-2157
Volume 2

Generic Environmental Impact Statement for Continued Storage of Spent Nuclear Fuel

Final Report
Public Comments

Office of Nuclear Material Safety and Safeguards

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NUREG-2157
Volume 2

Generic Environmental Impact Statement for Continued Storage of Spent Nuclear Fuel

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Public Comments

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Waste Confidence Directorate
Office of Nuclear Material Safety and Safeguards
U.S. Nuclear Regulatory Commission
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Abstract

This *Generic Environmental Impact Statement for Continued Storage of Spent Nuclear Fuel* (GEIS) generically determines the environmental impacts of continued storage, including those impacts identified in the remand by the Court of Appeals in the *New York v. NRC* decision, and provides a regulatory basis for a revision to 10 CFR 51.23 that addresses the environmental impacts of continued storage for use in future NRC environmental reviews. In this context, "the environmental impacts of continued storage" means those impacts that could occur as a result of the storage of spent nuclear fuel at at-reactor and away-from-reactor sites after a reactor's licensed life for operation and until a permanent repository becomes available. The GEIS evaluates potential environmental impacts to a broad range of resources. Cumulative impacts are also analyzed.

Because the timing of repository availability is uncertain, the GEIS analyzes potential environmental impacts over three possible timeframes: a short-term timeframe, which includes 60 years of continued storage after the end of a reactor's licensed life for operation; an additional 100-year timeframe (60 years plus 100 years) to address the potential for delay in repository availability; and a third, indefinite timeframe to address the possibility that a repository never becomes available. All potential impacts in each resource area are analyzed for each continued storage timeframe.

The GEIS contains several appendices that discuss specific topics of particular interest, including the technical feasibility of continued storage and repository availability as well as the two technical issues involved in the remand of *New York v. NRC*—spent fuel pool leaks and spent fuel pool fires. Finally the GEIS contains NRC's responses to public comments on the draft GEIS and proposed Rule and in doing so provides additional technical background on, and explanation of, the GEIS's analyses and conclusions.

The GEIS also discusses the NRC's Federal action—the adoption of a revised Rule, 10 CFR 51.23, to codify (i.e., adopt into regulation) the analysis in the GEIS of the environmental impacts of continued storage of spent fuel—and the options the NRC could take under the no-action alternative.

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D.2.6.5 – COMMENT: A commenter questioned whether money from the Nuclear Waste Fund was used for the Waste Confidence public meetings.

RESPONSE: Funding for the NRC's public meetings came from the NRC's general budget and did not come from the Nuclear Waste Fund. No changes were made to the GEIS or Rule as a result of this comment.

(889-3)

D.2.6.6 – COMMENT: A commenter noted the absence of a staff member from Congressman Issa's office at the Waste Confidence public meeting in Carlsbad, California, and questioned whether the NRC had any interactions with Congressman Issa regarding the rulemaking.

RESPONSE: The NRC keeps Congress fully and currently informed of the agency's regulatory activities. The NRC's Office of Congressional Affairs is the main conduit for NRC communications with Congress. Members of the Commission and NRC senior staff regularly work with the NRC's Office of Congressional Affairs to provide information to Congress and reply to inquiries from various committees of the House and the Senate and to Members of Congress who are interested in aspects of NRC responsibilities. No changes were made to the GEIS or Rule as a result of this comment.

(325-31-1)

D.2.7 Comments Concerning the Scope of the GEIS

D.2.7.1 – COMMENT: The NRC received comments asserting that the scope of the GEIS failed to include certain topics. One comment stated that the GEIS fails to address public safety and, because the scope is too narrow, the GEIS failed to effectively analyze the impacts on human health and the environment. The comment also claimed the GEIS failed to evaluate the indirect impacts of continued storage of spent fuel. Another commenter claimed that the NRC failed to address the environmental, political, and economic challenges associated with the continued production and accumulation of long-lived radioactive waste. The comment also stated that the NRC does not have a strategy for the long-term management of these wastes. Another comment indicated that, in general, not all of the framework for the management of spent fuel was in place. Two other comments asserted that the NRC should include the environmental impacts of spent fuel disposal within the GEIS. One comment claimed that the NRC is incorrect

to claim that the environmental impacts of spent fuel disposal are irrelevant to the GEIS. That same comment cited language used in the License Renewal GEIS that stated the environmental impacts of spent fuel disposal could not be resolved until the waste confidence EIS was completed. Another comment stated that by evaluating only the impacts of spent fuel storage, the NRC is excluding a major part of the nuclear fuel cycle. In particular, whether or not a repository is available will affect the impacts for both storage and disposal.

RESPONSE: The NRC disagrees with the comments. In the GEIS, the NRC has addressed the potential impacts to human health and the environment from the continued storage of spent fuel. For example, the GEIS includes evaluations of the impacts of normal releases and of accidents. The comment regarding indirect impacts provides insufficient information to understand what was expected in the GEIS. However, the GEIS does address indirect impacts, such as socioeconomic impacts (see, for example, the discussion in Section 4.2 of the GEIS). The long-term plan for the disposal of spent fuel is within the purview of the President and Congress; the current plan established by Congress is codified in the Nuclear Waste Policy Act (NWPA). While the NRC recognizes that the implementation of the current plan faces political and societal challenges, it is not the NRC's role to set policy in this area.

The impacts of disposal in a repository are addressed elsewhere in 10 CFR Part 51 and are beyond the scope of the current action. One comment argued that a footnote in the License Renewal GEIS implied that the Waste Confidence GEIS must address the impacts of spent fuel disposal. That interpretation of that footnote is incorrect. The footnote indicates that the work on the Rule was needed to provide information regarding the feasibility of a repository and the possible timing for the availability of a repository. This portion of the license renewal table was included in the proposed rulemaking *Federal Register* Notice (78 FR 37282, page 37322), and the NRC is including amendments to the license renewal table as part of the final rulemaking for this proceeding.

Finally, the GEIS specifically addresses how environmental impacts could be affected by delayed repository availability through the analysis of impacts in the three timeframes: short-term, long-term, and indefinite. No changes were made to the GEIS or Rule as a result of these comments.

(328-12-6) (714-1-23) (783-1-15) (897-7-12) (898-4-24)

D.2.7.2 – COMMENT: Several commenters suggested issues that they believe should have been within the scope of this GEIS. One commenter stated that the GEIS should have considered the complete safety of the environment and all living and interdependent organisms in the environment, economic impact to tax payers on prior mishandling, and a study of the industry and its liability for unforeseen consequences. Another commenter said the GEIS must consider various storage methods for spent fuel (e.g., hardened onsite storage [HOSS] and expedited transfer). The commenter also stated that the GEIS should have considered the

storage and transport of high-burnup fuel. One commenter requested that the NRC seek input from the public about the confidence in reclaiming "orphaned" sites and the fairness of living near storage facilities. Another commenter stated that the GEIS should be about more than just continued onsite storage, it should also more thoroughly address transportation, final disposition in a repository, and away-from-reactor storage. Another commenter stated that to satisfy EPA requirements, the GEIS should consider each reactor site separately and evaluate all costs involved. Another commenter stated that the NRC should consider alternatives for onsite and offsite storage of waste during and after the period of extended operation, offsite impacts during continued operation, long-term impacts and safety of the generation and storage of radioactive waste, comparative impacts of storage in pools versus dry storage, implications of storage on decommissioning, effects of storage and disposal if a repository is delayed, and alternatives and mitigations for these impacts. The commenter also stated that these issues are not generic and should be evaluated on a site-specific basis and the GEIS should therefore be tiered off of in subsequent site-specific EISs.

RESPONSE: The NRC agrees in part and disagrees in part with these comments. The scope of the GEIS is limited to the impacts of continued storage. Issues outside this scope, such as industry liability, selecting a specific dry storage approach (e.g., HOSS), and transportation outside of the period of continued storage are, therefore, not addressed in the GEIS. Additional information on other fuel storage options is provided in Section D.2.14.2 of this appendix. The evaluation in the GEIS addressed impacts to all resources that might be affected by the continued storage of spent fuel. With regard to spent fuel burnup, Chapter 2 of the GEIS explains that for purposes of environmental impact analysis, the NRC relies on the larger lifetime amount of spent fuel discharged at low burnups. However, as discussed in Section D.2.16.13 of this appendix, information on the characteristics of low-burnup, high-burnup and mixed-oxide (MOX) fuels has been added to the GEIS in Appendix I to help clarify the similarities between these fuel types. Regarding site-specific analyses, this document is, by its nature, a generic evaluation. Prior to the completion of an individual licensing action, the NRC will conduct a site-specific environmental review and document the results of this review in an EA and FONSI or EIS. For more information regarding the generic evaluation of impacts, see Section D.2.11.1 of this appendix.

Comments that address activities that occur during plant operations (including the generation of the waste) are outside the scope of the GEIS and Rule.

The NRC included the impacts of both pool and dry cask storage in its evaluation, but did not compare them to each other in the GEIS. As discussed in the GEIS, a spent fuel pool will already exist at each site and its impacts to the environment in the period of continued storage are minor. The NRC assumes that all fuel will have been moved from the pool to casks by the end of the short-term timeframe. Therefore, both wet and dry storage are a necessary part of

the storage of the fuel during continued storage and a comparison of them is not necessary. For additional information, see Section D.2.38.10 of this appendix.

The GEIS addresses the cumulative impacts of continued storage and decommissioning (in Chapter 6) and the impacts of a repository being delayed or not available (Chapters 4 and 5). For additional information regarding decommissioning, see Section D.2.43.1 of this appendix. Alternatives are discussed in Section 1.6 of the GEIS—these are (as NEPA requires) alternatives to the proposed action (revising the Rule). In general, the NRC will address mitigation in site-specific licensing reviews. Section D.2.11.1 of this appendix provides additional insight into this approach. However, the GEIS does discuss mitigation related to aging, damaged, or degraded fuel (see Section 2.2.2.1 of the GEIS and Section D.2.17.4 of this appendix for additional information regarding this issue). Any determinations by the NRC about whether to require mitigation measures of any type will occur on a site-specific basis during facility licensing or during the course of ongoing NRC oversight. No changes were made to the GEIS or Rule as a result of these comments.

(11-2) (30-21-3) (45-11-11) (112-28-3) (246-32-2) (611-19) (693-1-11) (706-1-15)

D.2.7.3 – COMMENT: The NRC received a comment requesting the NRC clarify that ESPs are not included within the scope of the GEIS and Rule. The commenter stated that because ESPs do not authorize the generation or storage of spent fuel, NEPA does not require consideration of the environmental impacts of continued storage of spent fuel for ESP applications. The commenter also provided numerous conforming revisions to the GEIS to clarify that waste confidence does not apply to ESPs, in particular the cost-benefit analysis in Chapter 7 and in Appendix H.

RESPONSE: The NRC agrees with the comment that clarification of the scope of the proposed Rule change is appropriate. The NRC recognizes that neither the current language of the Rule, nor the proposed revision to that section, expressly addresses whether the Rule applies to ESP reviews.

However, the NRC disagrees with the comment that ESPs are not covered by the Rule. The clear purpose of the regulation was to preclude the need for a site-specific analysis of the environmental impacts of continued storage for all power-reactor-related and ISFSI-related licensing actions, including spent fuel generated by new reactors. This purpose is evident from the Commission's intention in past Waste Confidence proceedings and the 2007 rulemaking on Part 52 to encompass waste produced by a new generation of reactors—including those licensed under the Part 52 regime, which includes ESPs (49 FR 34688; 55 FR 38472; and 72 FR 49352). That the regulation did not expressly include ESPs in the list of reactor licensing actions under Parts 50 and 52 for which a site-specific analysis of the environmental impacts of continued storage is not necessary, coupled with the absence of an explanation as to why ESPs were not included in the regulation, is evidence that this was an oversight on the part of the NRC. Not including ESPs would also lead to the anomalous result—again, unexplained by NRC—of

RESPONSE: The NRC agrees with the comment that there might be other options available in the future to meet the same objectives as having a DTS at each spent fuel storage installation. The GEIS assumed a DTS at each storage site as a conservative assumption for the purpose of evaluating potential environmental impacts of continued storage. As with all NEPA analyses, the assumptions in the GEIS in no way approve actions or constitute requirements. No changes were made to the GEIS or Rule as a result of this comment.

(827-2-1)

D.2.17.3 – COMMENT: Several commenters stated that there will be unspecified difficulties, costs, spills, and accidents stemming from transfers of spent fuel from spent fuel pools to dry casks, and from dry casks to other dry casks. One commenter stated that there may not be room on the existing sites to construct the necessary DTSs and ISFSIs. In addition, one commenter asserted that no generic environmental impacts assessment can be made because of site-specific variations in the condition of spent fuel pools, canisters, and casks; the existence of multiple types of dry storage systems; and the unverified performance of the reference DTS. Another commenter asserted that the GEIS discussion of effluent radiation monitoring is an admission that there will be radiological releases from the DTSs over time. One commenter expressed general skepticism about the reliability of the NRC's DTS and dry cask assumptions because the NRC's assessments of the technical capabilities of dry casks "keep expanding and improving as time progresses and the prospect of an available repository diminishes."

RESPONSE: The NRC disagrees with the comments. Because continued storage activities involving a DTS are assumed to occur in the long-term timeframe after the operating license of a power reactor expires, the DTS activities evaluated in the GEIS would occur many decades into the future (i.e., beyond 60 years past the term of the operating license). Therefore, some uncertainty exists regarding the specific methods and equipment that would be used. For the purpose of evaluating environmental impacts in the GEIS, the NRC conservatively assumed DTSs would be employed based on existing technology and regulations. This assumption is conservative because constructing, operating, and replacing DTS facilities would have greater environmental impacts than other plausible future options for addressing at-reactor transfer needs (e.g., use of overpacks that would not require bare fuel handling). In addition, industry has decades of operating experience with wet transfer of new fuel and spent fuel, which involves some spent fuel handling equipment and procedures similar to what would be used in a DTS. Based on these factors, the NRC considers the assumption regarding the future use of DTSs to be reasonable. Additional details about the design, operation, and safety of the DTS concept are provided in the supporting references in Sections 2.1.4 and 2.2.2.1 of the GEIS.

While spent fuel transfer operations can present challenges to operators (e.g., working with damaged fuel [see Section D.2.17.4 of this appendix for more information]), as described in Section 4.17.2 of the GEIS operation of a DTS would be similar to the operations conducted at current reactor sites with licensed ISFSIs where spent fuel is loaded in dry storage cask

In response to comments about damaged fuel, NRC has added information within Section 2.2.2.1 of the GEIS describing damaged fuel in the context of DTS operations, including descriptions of methods for handling damaged spent fuel. No other changes were made to GEIS and no changes were made to the Rule as a result of these comments.

(2-4) (163-22-7) (230-5) (336-8) (377-5-18) (431-10) (608-18) (819-14) (867-3-13) (867-2-20) (898-2-19) (919-4-11)

D.2.18 Comments Concerning GEIS Assumptions – Timeframes

D.2.18.1 – COMMENT: Many commenters provided comments on the likelihood of the indefinite timeframe and the NRC's statement that the short-term timeframe is the most likely timeframe. Commenters questioned the NRC's statements in the draft GEIS that the indefinite timeframe was highly unlikely and stated that it was unreasonable to assume a repository would be available within the short-term timeframe.

In contrast, other commenters expressed support for repository availability in the short-term timeframe. One commenter stated a no-repository scenario is contrary to current law and is remote and speculative and represents a worst case, which is not required by NEPA.

RESPONSE: Geologic disposal remains the national strategy for the disposition of spent fuel under the NWPA and the Federal government, through the DOE, is continuing its work on a disposal solution for spent fuel. Based on these factors and the technical feasibility of a geologic repository (discussed in Appendix B of this GEIS), the NRC has concluded that siting, constructing, and licensing of a repository within the short-term timeframe is the most likely outcome. Consequently, the NRC believes that the indefinite timeframe is the least likely of the three timeframes. However, sufficient uncertainty remains in the timing of the effort to open a repository that the NRC cannot completely rule out the possibility that a repository will not be available by the end of the short-term timeframe. Therefore, the NRC has prepared an analysis of an additional 100 years of continued storage (i.e., the long-term timeframe) and, in accordance with the direction of the Court of Appeals, has assumed that a repository never becomes available (i.e., the indefinite timeframe).

In addition, a number of comments were submitted that expressed concern regarding both the costs and responsibilities of continued storage (see Section D.2.42 in this appendix). DOE has estimated that future liabilities, should the U.S. Government not take custody of spent fuel, will total about $20.7 billion through 2020 and may cost about $500 million each year after that (GAO-12-797, GAO 2012). Furthermore, the NRC acknowledges that, because of delays in the siting and licensing of a repository, the Federal government bears an increasing financial responsibility for spent fuel storage costs, and it may become responsible for paying all the costs associated with spent fuel storage at some time in the future. Financial liabilities of this magnitude support the NRC's view that the short-term timeframe is the most likely outcome.

Appendix B in the GEIS has been revised to provide further clarification of the basis for the Commission's conclusions concerning the feasibility of geologic disposal. No changes were made to the Rule as a result of these comments.

(59-10) (112-20-1) (163-7-3) (208-2) (222-13) (244-11-6) (250-7-4) (431-5) (459-7) (532-6) (544-23) (544-5) (556-2-7) (611-25) (714-1-10) (818-1) (827-2-4) (827-2-5) (827-5-6) (919-4-8) (942-9)

D.2.18.2 – COMMENT: Commenters expressed concerns regarding the adequacy of the evaluation of future impacts in the GEIS. Although some commenters questioned the credibility of the estimates of future impacts for the short-term timeframe, the majority of comments expressed concern regarding the long-term and indefinite timeframes. These concerns were mainly attributed to uncertainty in how conditions may evolve in the future. Commenters stated that the impacts, including costs, during the indefinite period need to be analyzed, and that, based on the impact determinations of SMALL in the draft GEIS, the analysis of costs appears inadequate. In contrast, one commenter stated that with proper maintenance and monitoring spent fuel could be indefinitely stored in pools or dry casks.

RESPONSE: The NRC agrees that evaluation of future environmental impacts are uncertain due to uncertainties in future conditions, however, the presence of uncertainty does not invalidate nor preclude the development of reasonable determinations of potential environmental impacts in the GEIS. Section 1.8.3 of the GEIS presents assumptions used for evaluating environmental impacts that provided appropriate and reasonable bounds for projecting future conditions and activities related to continued storage (e.g., see response to comments in Sections D.2.18.8 and D.2.19.1 of this appendix.

The NRC does not agree that the adequacy of the GEIS should be based on the impact determinations being SMALL. The GEIS fully describes the evaluations and impact determinations for each resource area and each timeframe. The NRC has responded to comments for each resource area, including postulated accidents and climate change, made any necessary changes to the GEIS, and determined the GEIS evaluations are appropriate (see Chapters 4 and 5 for at-reactor and away-from reactor storage impacts.

The suggestion that costs of continued storage be considered in the GEIS is addressed in Section D.2.18.1 of this appendix. Except for the changes made to the GEIS discussed in Section D.2.42.1, no changes were made to the GEIS or Rule as a result of these comments.

(112-20-2) (163-39-1) (163-12-2) (163-24-2) (163-16-7) (208-1) (208-3) (239-2) (245-14-4) (250-17-1) (250-17-2) (250-26-2) (250-69-3) (250-9-3) (250-5-4) (250-18-5) (262-4) (326-21-3) (326-53-4) (341-1-16) (341-1-20) (373-10) (402-3) (417-10) (431-7) (552-1-25) (553-1) (652-2) (674-5) (701-4) (714-1-1) (805-1) (823-76) (823-77) (860-3) (897-4-1) (919-2-1) (919-2-3)

D.2.18.3 – COMMENT: Many commenters stated that the GEIS timeframes are too long or expressed confusion about them. Some commenters suggested that the GEIS use shorter timeframes and not include the consideration of long-term storage. Other commenters considered long storage times to be permanent storage or "de-facto" disposal, which they assert is contrary to the NWPA. Some commenters also expressed support for the reasonableness of the timeframes.

One comment suggested that the NRC add text to Section 1.8.2 of the GEIS to clarify that the timeframes presented are just one analytical approach that ensures all spent fuel is analyzed for the entire period before geologic disposal and that other analytical approaches would have worked just as well.

RESPONSE: The NRC acknowledges the comments in support of the timeframes selected, and agrees with the comment that NRC could have used different analytical approaches (i.e., different time periods) to analyze the environmental impacts of continued storage. However, the NRC disagrees that consideration of long-term storage should not be included in the GEIS. Regardless of the number and length of specific timeframes, the GEIS needs to evaluate and disclose the impacts of continued storage. However, sufficient uncertainty remains in the timing of the effort to open a repository that the NRC cannot completely rule out the possibility that a repository will not be available by the end of the short-term timeframe. The NRC has therefore prepared an analysis of an additional 100 years of continued storage (the long-term timeframe) and, in accordance with the direction of the Court of Appeals, has analyzed the indefinite timeframe.

The timeframes selected for the GEIS conform to the GEIS assumption that dry cask storage systems would be replaced every 100 years. The NRC believes the replacement period provides reasonable increments of time for evaluating environmental impacts because the replacement of dry cask storage systems is likely to be more environmentally significant than routine storage operations. In addition, replacement activities provide a distinct period of time to analyze. Although the GEIS evaluates the impacts of storage activities for all three timeframes, it does not authorize storage during these timeframes. Authorization for storage, if it were ultimately pursued, would require separate licensing actions with requisite environmental analysis.

The NRC notes that some comments expressed confusion regarding the timeframes and their relationship to licensed facility life. As explained in the GEIS, including Section 1.8.2, the environmental impacts considered in the GEIS are for the time period "after" the licensed life for reactor operations. At that time, the licensee would no longer be authorized to operate a reactor, but would continue to store spent fuel onsite under either its 10 CFR Part 50 or Part 52 license, or a 10 CFR Part 72 license. During this time, the licensee would remain under the NRC's regulatory control and NRC inspections and oversight of storage facilities would continue. The NRC monitors the performance of ISFSIs (at both decommissioned and

shutdown reactor sites and operating reactor sites) by conducting periodic inspections. As discussed in Section D.2.18.4, revisions to Section 1.8.2 of the GEIS have been made to provide further context for the evaluation of the impacts after licensed life that also provide clarity for the timeframes considered. No changes were made to the Rule as a result of these comments.

(219-8) (262-6) (326-43-1) (447-2-20) (447-2-3) (544-14) (544-20) (544-24) (544-29) (622-1-11) (622-1-14) (637-8) (646-17) (689-3) (698-1) (819-2) (836-52) (919-7-20) (919-6-6) (930-3-4)

D.2.18.4 – COMMENT: Some commenters stated that the timeframes in the GEIS, which begin at the date the plant ceases operations, do not account for casks that have been loaded and are sitting for years prior to the cessation of plant operations. Commenters requested that the consideration of storage run from the date that the spent fuel is put into a cask, not the date that the plant ceases operations. One commenter noted that the 100-year timeframe for cask replacement does not take into consideration any information from the manufacturer of the casks, such as a warranty or statement on the useful life of the cask.

One commenter stated that the NRC should delete any reference to the storage timeframe including operations of the plant (i.e., text starting on page 1-12 and continuing to 1-13 in Section 1.8.2 of the draft GEIS) to make clear that continued operation of the plant is separate and distinct from storage times for dry casks.

RESPONSE: The NRC agrees in part and disagrees in part with the comments. In general, the NRC agrees with commenters that the consideration of continued storage needs to consider the age of storage facilities in place at the beginning of the continued storage period; however, the NRC disagrees with the assertions that the environmental impacts of storage of spent fuel during reactor operations should be included in the GEIS.

Prior to the completion of an individual licensing action (e.g., review for a combined license), the NRC will conduct a site-specific environmental review and document the results of this review in an EA and FONSI or EIS. The environmental impacts of storing spent fuel at reactor facilities during the licensed life for reactor operations will be evaluated during that review. Though those impacts are assessed separately, they will be considered in conjunction with the impacts in the GEIS at the time of licensing, and thus do not need to be considered as part of the GEIS.

As explained in the GEIS, including Section 1.8.2, the environmental impacts considered in the GEIS are for the time period after the licensed life for reactor operations and the age of the storage facilities are considered in the analysis. For example, the GEIS assumption that replacement of the entire ISFSI would occur over the course of each 100-year interval, starting at the beginning of the long-term storage timeframe (see Section 1.8.3 of the GEIS) specifies the beginning of the long-term storage timeframe because: (1) a typical spent fuel pool reaches its licensed capacity limit about 30 years into the licensed life for operation of the reactor after

which some spent fuel would need to be removed from the spent fuel pool and transferred into a dry cask storage system; (2) for a reactor that is assumed to operate for 80 years the spent fuel that was first placed in dry casks would have been stored on the order of 50 years at the beginning of the short-term timeframe; and (3) the beginning of the long-term timeframe, which occurs 60 years after the end of reactor operations, represents a period of approximately 100 years of dry cask storage for the spent fuel that was initially placed into dry cask storage, which is the time period over which it is assumed a dry cask storage system would be replaced. Thus, the consideration of replacement of dry cask storage systems at the beginning of the long-term storage timeframe explicitly accounts for the assumed lifetime of the dry cask storage system. The NRC has revised Sections 1.8.1, 1.8.2, and 1.8.3 of the GEIS to clarify the approach in the GEIS for evaluation of cask lifetimes in the context of evaluating impacts after licensed life for reactor operations. No changes were made to the Rule as a result of these comments.

The consideration of "warranty" type information, as suggested by the commenter, is not expected to add further significant information beyond what has already been considered for estimating the behavior and longevity of dry cask storage systems. Appendix B of the GEIS describes the design of storage casks as well as national and international experience with storage casks in support of the longevity of dry casks (e.g., current understanding for slow degradation rates of dry storage casks). The GEIS assumes casks will be replaced every 100 years as a conservative assumption to facilitate the NRC's environmental analysis. For example, this assumption results in increased land use and generation of concrete waste. The NRC notes that the 100-year replacement interval is not intended to convey that dry casks and facilities need to be replaced every 100 years to maintain safe storage. The NRC considered experience with dry cask storage systems, information related to certification and regulatory oversight of dry cask storage systems, and monitoring and maintenance of dry cask storage systems to provide an informed basis for understanding the behavior of dry cask storage systems and estimating a replacement interval for the GEIS that is considered conservative (i.e., replacement times would most likely be longer than 100 years).

(328-2-4) (417-1) (783-1-5) (783-2-5) (836-35) (930-2-9)

D.2.18.5 – COMMENT: Commenters expressed concern that the assumptions in the GEIS regarding the longevity of storage casks and pools are based on NRC experience with spent fuel storage for shorter durations than the lifetimes of up to 140 years for spent fuel pools and 100 years for spent fuel casks projected in the GEIS.

RESPONSE: The NRC agrees in part and disagrees in part with these comments. The NRC agrees that spent fuel is currently stored in spent fuel pools and dry casks for less time than the NRC assumes could occur in the GEIS (i.e., the GEIS assumes a spent fuel pool is operational for up to 140 years and dry casks are in service for 100 years). The NRC disagrees with the comments' concerns that the experience with spent fuel storage does not support the storage times considered in the GEIS because the assumed GEIS storage times are longer than the current storage duration.

Appendix B of the GEIS provides the technical basis for the NRC's conclusions that it is feasible that spent fuel may be safely stored in spent fuel pools and dry casks for the periods projected in the GEIS. This analysis in Appendix B includes support for the robust structural design and construction of spent fuel pools and dry cask storage, their slow rate of degradation, and programs for monitoring and maintenance at storage facilities. In response to public comments, the NRC has revised Appendix B to add additional information regarding the role of monitoring and maintenance programs for collecting operational experience. No changes were made to the Rule as a result of these comments.

(544-13) (783-3-21) (783-2-7) (867-2-4) (920-24)

D.2.18.6 – COMMENT: One commenter suggested that the GEIS could be developed around scenarios that present the expected impacts on the environment at each of the timeframes used in the GEIS.

RESPONSE: The NRC agrees that the GEIS could present the environmental impacts using a 'scenario' approach as suggested by the comment. However, the NRC has decided to use its well-established format for EISs; the GEIS is organized to present the environmental impacts for each timeframe according to the specific resource areas of the affected environment. The comment did not suggest the GEIS approach was inappropriate. No changes were made to the GEIS or Rule as a result of this comment.

(867-1-6)

D.2.18.7 – COMMENT: One commenter raised concerns with quality assurance violations related to the design and manufacture of dry casks and questioned how the NRC could have confidence in indefinite dry cask storage when significant quality assurance issues exist.

RESPONSE: The NRC disagrees with the comment's assertion that past quality assurance issues undermine confidence in the safety of dry cask storage systems. Although there have been isolated instances involving dry cask storage system design or operational issues, the extent of the issues identified have not called into question the safety of dry cask storage systems. Further information on monitoring and maintenance of dry cask storage systems is provided in Section D.2.38.19. No changes were made to the GEIS or Rule as a result of this comment.

(327-10-3)

D.2.18.8 – COMMENT: Commenters questioned the technical and factual basis for the assumptions that (1) pool storage would end 60 years after the licensed life for operation of the reactor, (2) storage facilities (i.e., dry casks) would be replaced every 100 years, and (3) the amount of spent fuel considered in a timeframe was appropriate. Some commenters stated that

storage assumptions need to be based on regulatory requirements, others asserted that experience did not challenge the GEIS assumptions.

RESPONSE: The NRC agrees in part and disagrees in part with the comments. The NRC agrees that assumptions made in the GEIS for evaluating environmental consequences are not always based on regulatory requirements; however, the NRC disagrees with statements in the comments that the assumptions in the GEIS need to be based on regulatory requirements.

The NRC has made reasonable assumptions that support the analysis of the environmental impacts of continued storage in the GEIS. The cessation of pool storage 60 years after the licensed life of the reactor is reasonable because (1) there is no need to cool spent fuel in a pool for more than 60 years after a reactor stops operating; (2) operational costs associated with pool storage exceed dry cask storage costs; and (3) experience with decommissioning of nuclear power plants indicates that spent fuel pools are decommissioned before the end of the 60-year period. No dry cask storage systems have reached a 100-year service time; however, current information supports low degradation rates for dry cask storage systems (see Appendix B of the GEIS and Sections D.2.38.5 and D.2.38.19 of this appendix for further details).

The NRC is not aware of information that would suggest that dry cask storage systems would need to be replaced after 100 years of service. However, the NRC believes that the 100-year replacement period provides a reasonable timeframe for the routine replacement of dry storage systems, and that actual storage facility replacement will be needed less frequently than assumed in the GEIS. The conservative nature of this assumption ensures that the environmental impact determinations in the GEIS are unlikely to underestimate the actual environmental impacts, should continued storage be necessary.

The GEIS considers the environmental impacts from continued storage for an at-reactor site and an away-from-reactor ISFSI. The amount of spent fuel considered for each site is consistent with the operational volume of a single facility (i.e., 1,600 metric tons of uranium [MTU] of spent fuel for the at-reactor site and 40,000 MTU for an away-from-reactor ISFSI). No changes were made to the GEIS or Rule as a result of these comments.

(163-2-4) (163-1-5) (200-3) (244-14-3) (244-14-4) (473-10-1) (473-12-11) (473-12-14) (473-11-2) (473-17-2) (473-12-8) (473-12-9) (556-2-4) (608-14) (637-6) (669-10) (916-2-1) (916-2-2) (919-3-5)

D.2.19 Comments Concerning GEIS Assumptions – Institutional Controls

D.2.19.1 – COMMENT: Many commenters questioned the reasonableness of the draft GEIS assumption that effective institutional controls will continue indefinitely into the future. Some of these commenters argued that the NRC could not support a conclusion that loss of institutional controls is remote and speculative. Other commenters believe that the NRC's conclusions

Another commenter stated that the NRC did not consider a 2012 flooding event at the Fort Calhoun nuclear power plant, in which the Missouri River nearly reached the grade level of the onsite ISFSI. The commenter stated that the NRC has no means to assure that this would be the case in future catastrophic flooding; or that Cooper Nuclear Station or Oconee Nuclear Station dry casks would be above flood level after a catastrophic upstream dam failure. The commenter stated that should ISFSI flooding occur, the lower cooling vents of the casks could potentially become submerged or clogged with debris that licensee personnel would not be able to intervene to clear, resulting in a possible cask rupture.

Another commenter stated that the GEIS should have considered criticality accidents in dry casks storage systems and cited NUREG/CR–6835, *Effects of Fuel Failure on Criticality Safety and Radiation Dose for Spent Fuel Casks* (Elam et al. 2003), in support of the comment. Other commenters stated that dry cask storage systems offer significant robust protection from natural phenomena.

Some commenters stated that the accident in March 2011 at the Fukushima Dai-ichi nuclear power plant demonstrates that dry cask storage systems are more robust than spent fuel pools. Commenters described effects of the August 23, 2011 Mineral, Virginia earthquake on spent fuel storage facilities at the North Anna Nuclear Generating Station and stated that seismographs have been removed from at least one plant. A few commenters noted that the heavy cement dry storage casks moved 1 to 4 in. One commenter went on to point out that, there are earthquake faults near specific plants and that there is no proof that, after an earthquake, the waste will be able to be removed safely because no fuel has been removed from dry cask storage after a long period of time. Some commenters stated that the accident at Fukushima Dai-ichi and the Mineral, Virginia earthquake demonstrate that dry cask storage systems are robust, one commenter stated dry cask storage systems are more robust than spent fuel pools.

RESPONSE: The NRC agrees in part and disagrees in part with the comments. The NRC agrees that dry cask storage systems offer significant robust protection from natural phenomena, as evidenced by the continued safety of storage facilities after earthquakes struck spent fuel storage facilities at the Fukushima Dai-ichi and North Anna nuclear power plants. However, the NRC disagrees that the analysis in the GEIS is deficient with respect to the analysis of accidents involving dry cask storage systems and disagrees that the additional information and documents cited by the commenters affects the analysis or conclusions in the GEIS.

The NRC does not agree with the comments' statements about the risk of spent fuel fires in dry cask storage systems. Dry cask storage systems are passive and robust engineered structures designed to withstand natural forces, such as earthquakes and wind-borne missiles. There are no active safety systems that could fail and thereby increase the likelihood of a release of radioactive material. Even in the highly unlikely event of an accidental breach of a cask, or cask

(388-2) (391-1) (398-2) (399-2) (400-2) (408-1) (448-1) (466-2) (538-1) (548-10) (549-3) (557-1) (574-1) (592-1) (601-3) (637-12) (642-3) (674-4) (675-1) (682-2) (683-2) (694-3-13) (694-2-23) (694-2-26) (694-2-28) (694-1-6) (694-1-7) (697-1-7) (697-2-8) (745-3) (753-2) (808-1) (812-1) (827-2-3) (827-3-8) (863-3) (864-10) (885-1) (886-1) (909-1) (911-1) (942-8) (949-1) (949-3) (949-5) (949-6)

D.2.38.3 – COMMENT: One commenter suggested that the GEIS include specific information on aging management. The commenter suggested that the GEIS include information on how the aging management program will provide for monitoring the integrity of dry storage system components and the potential emissions specific to dry storage systems during the 100-year storage timeframe.

RESPONSE: The NRC agrees that an aging management program is an important component of the NRC's regulatory oversight of spent fuel storage. Applicants for specific licenses (10 CFR 72.42, Issuance of license) and CoC renewals (10 CFR 72.240, Conditions for spent fuel storage cask renewal) are required to describe a program for the management of issues associated with aging that could adversely affect structures, systems, and components important to safety; structures, systems, and components that are necessary to fulfill a function that is important to safety; or support the function of a structure, system, or component that is important to safety. The NRC conducts a review of the aging management activities described in these applications. The NRC will only approve of the renewal application if the program is adequate to provide reasonable assurance that aging effects would be managed during the period of extended operation.

All ISFSI sites are required to meet the dose limits in 10 CFR 72.104, Criteria for radioactive materials in effluents and direct radiation from an ISFSI or MRS. Appropriate radiation monitoring and the aging management program requirements will be reflected in the terms, conditions, and technical specifications of the renewed CoC and thus made applicable to the general licensee per 10 CFR 72.212(b). For specific licenses, radiation monitoring and aging management program requirements will be reflected in the terms and conditions of the renewed specific license. The NRC will monitor the general or specific licensee's compliance with the terms and conditions of their license through the NRC's inspection program. Guidance on aging management programs is available in Chapter 3 of NUREG–1927, Standard Review Plan for Renewal of Spent Fuel Dry Cask Storage System License and Certificates of Compliance (NRC 2011g).

Additional information on the aging management program has been added to Appendix B of the GEIS. No changes were made to the Rule as a result of this comment.

(915-10)

Regarding costs of storage, utilities are responsible for the cost of storage, including the cost of purchasing dry casks. However, in response to high public interest about costs, the NRC has included some cost information regarding continued storage in Chapter 2 of the GEIS. No changes were made to the Rule or GEIS as a result of these comments.

(112-30-2) (116-3) (163-6-1) (326-4-4) (326-35-5) (473-8-7) (484-5) (556-5-3) (671-1) (778-3) (929-15) (929-7)

D.2.38.5 – COMMENT: Several commenters expressed concern about spent fuel storage in dry storage casks.

Commenters stated that casks need to withstand terrorism, tornadoes, floods, airplane crashes, underwater submersion, and severe earthquakes. To support their belief that spent fuel cannot be stored safely in dry storage casks for hundreds of years, commenters provided examples of past issues with casks (e.g., fabrication, cracking, corrosion, welds, seals, loading and unloading, leaking, clogged air flow vents, equipment failure, concrete storage pads, location of storage pads, hydrogen ignition incidents, a failed dry cask test, and quality control and assurance) that they believe indicate that spent fuel cannot be stored safely in dry storage casks for hundreds of years. One commenter stated that NRC had allowed manufacturers to build casks before issuance of the CoC, and another commenter stated that the NRC has exempted defective casks in the past. Commenters questioned how the NRC can have confidence in safe dry storage forever given the many documented issues and data gaps associated with cask storage. Several commenters expressed concern about storage of high-burnup fuel in dry casks due to the limited experience with this fuel.

A few commenters noted the maximum cask life (alternatively described as 300 years, 50 years, or 20 years) is not sufficient for indefinite storage. Commenters asked how the NRC will assure that the casks are replaced after their lifetime or earlier if the cask leaks and who would pay for any replacement. Commenters stated that there is no proof that spent fuel can be safely removed from casks or transported in the future as no cask has ever been unloaded.

However, other commenters stated that casks can withstand environmental disasters as evidenced by Fukushima. Commenters noted that casks have been used safely for decades and systems have become more robust over time. Commenters noted that dry cask storage systems will continue to evolve in the future with enhancements that will improve safety. Commenters provided some examples of improvements that have been made to the casks, including higher capacity casks that reduce handling activities. Some commenters encouraged the NRC to make dry cask storage safer.

One commenter was concerned that the NRC's cask certification process occurs too quickly and locks out public involvement. The commenter stated that the process has been taken over by the industry and that the lack of rigorous oversight has resulted in a lack of cask design field

testing. Commenters noted that the casks are not approved for geologic disposal and requested information on when final geologic disposal casks would be available.

Commenters encouraged the NRC to comprehensively evaluate and validate the sufficiency of the design life of storage systems.

One commenter submitted a paper for a method to take advantage of the heat that comes off the dry storage casks.

RESPONSE: The NRC agrees in part and disagrees in part with the comments. The NRC agrees with the comments that casks have been used safely for decades and that the designs continue to evolve. The NRC disagrees with the comments that indicate spent fuel cannot be stored safely in dry casks. Appendix B of the GEIS analyzes the feasibility of safe storage of spent fuel in casks. The analysis describes proven storage methodologies, practical operating experience, and the regulatory oversight provided by the current regulatory framework, allowing the NRC to determine that spent fuel can continue to be stored safely in the short-term timeframe with only routine maintenance and in the long-term and indefinite timeframes with cask replacement every 100 years.

The NRC assures safety by requiring multiple layers of protection against radiation releases. The storage casks provide an important barrier and the fuel cladding provides another layer of protection. The design requirements imposed by regulation ensure that the casks will maintain shielding, confinement, and subcriticality during normal and off-normal conditions of storage, postulated accidents, and natural events. The NRC reviews each application for a cask CoC to determine whether the storage cask design meets the requirements at 10 CFR Part 72. As part of this review, the NRC performs confirmatory analysis to verify the information in the application. The CoC application and amendment review processes are thorough, and the information submitted by the applicants, the NRC questions (e.g., requests for additional information and requests for supplementary information), and the applicant responses are available for public review during the processes (with some information redacted for security or proprietary reasons). As part of the review, the NRC evaluates the applicant's QA program to ensure it meets the requirements in 10 CFR Part 72.

Storage cask performance is evaluated against a range of normal and off-normal conditions, accidents, and external events. For normal conditions of storage, the casks are evaluated for maximum high and minimum low ambient temperatures and must simultaneously include the effects of solar insolation. In addition, the NRC evaluates the operational environment that a cask will experience when it is being loaded, prepared for storage, and transferred to the storage pad. The evaluations for off-normal conditions include variations in temperatures beyond normal, failure of 10 percent of the fuel rods combined with off-normal temperatures, failure of one of the confinement boundaries, partial blockage of air vents, out-of-tolerance equipment performance, equipment failure, and instrumentation failure or faulty calibration. The

applicant is required to evaluate the storage cask for a cask drop and tipover, fire, fuel rod rupture, and air flow blockage (for vented storage casks). In addition to accident conditions, the following natural phenomena are evaluated: flood, tornado, earthquake, burial under debris, lightning strike, and other phenomena (e.g., seiches, tsunamis, and hurricanes), as appropriate, depending on the storage cask location. The Commission has determined that evaluation of terrorist strikes and large plane impacts, on the other hand, are beyond-design-basis events that do not need to be evaluated by an applicant for a license or CoC. The test (at Aberdeen Proving Ground) referenced in the comments, discusses perforation of a cask by an armor piercing missile, which is also a beyond-design-basis event analogous to a terrorist strike.

Once the NRC review is completed, the cask design is approved by rulemaking, which provides the public an opportunity to comment. The NRC review process is similar for site-specific ISFSI license applications and amendments under 10 CFR Part 72, but licensing ISFSIs includes the opportunity for the public to request a hearing.

The NRC performs regular inspections at the cask fabrication facilities for CoCs, at both specifically and generally licensed ISFSI sites. The NRC also performs regular inspections of CoCs and license holders’ QA programs.

With respect to the comment on early fabrication of casks and exemptions, the NRC does allow early fabrication and exemption requests. According to NRC regulations at 10 CFR 72.234(c), “[a]n applicant for a CoC may begin fabrication of spent fuel storage casks before the Commission issues a CoC for the cask; however, applicants who begin fabrication of casks without a CoC do so at their own risk. A cask fabricated before the CoC is issued shall be made to conform to the issued CoC before being placed in service or before spent fuel is loaded.” If a storage cask does not meet the design approved by the NRC, then the CoC holder must either repair the non-conforming part, perform an evaluation under 10 CFR 72.48 to determine whether the CoC holder can deviate from the CoC without prior NRC approval, or obtain approval by the NRC to use the non-conforming part. Exemption applications submitted under 10 CFR 72.7 are reviewed by the NRC to ensure that they are authorized by law and will not endanger life or property or the common defense and security and are otherwise in the public interest. The NRC completes both a safety review and an environmental review before approving any exemptions.

The NRC follows up on potential safety issues with casks through the inspection program or, in some cases, through the allegation process. When concerns or issues are substantiated, the NRC takes appropriate follow-up actions for those with a resulting safety or regulatory concern. The NRC has previously addressed, or is addressing, the various concerns raised in the comments, many of which were identified by the NRC. For example, the NRC identified the potential for chloride-induced stress-corrosion cracking of austenitic stainless steel in Information Notice 2012-20 (NRC 2012l).

The NRC requires its licensees to implement monitoring and surveillance programs and licensees must take the necessary actions to ensure that the necessary integrity of required systems and components is maintained (see 10 CFR 72.44(c)). Surveillance programs include periodic inspections of storage cask vents to ensure that debris does not block the vents.

While it is true that no storage cask has been unloaded at an ISFSI, casks have been unloaded at national laboratories. Applicants provide unloading procedures as part of the CoC application or site-specific application and the NRC reviews the procedures as part of its review.

With respect to the comments on transportation, some storage casks have been certified for both storage and transportation. Spent fuel has been safely transported for over 30 years with no incidents or releases of radiation.

With respect to storage term, existing casks are certified for either 20 or 40 years, and can be renewed. One of the issues examined for renewals is storage of high-burnup fuel. 10 CFR Part 72 contains both cladding integrity and retrievability requirements. For CoC renewal applicants, the NRC would consider any degradation mechanisms associated with the fuel pellet itself or the cladding that could challenge the cladding integrity and retrievability of fuel from storage. Hydride reorientation that may occur in high-burnup fuel during storage, which could embrittle the cladding at lower long-term temperatures, is predominately a transportation issue. Under renewed CoCs, licensees are required to manage any effects associated with this degradation if it could adversely affect structures, systems, and components important to safety. The NRC believes sufficient data are available to project that high-burnup fuel can be safely stored and retrieved. Licensees must include aging management programs in their renewal applications to manage issues associated with aging that could adversely affect structures, systems, and components important to safety (including corrosion). Licensees must also include time-limited aging analyses that demonstrate that structures, systems, and components important to safety will continue to perform their intended functions for the requested period of extended operation. For more information on high-burnup fuel, see Appendix I of the GEIS and Section D.2.38.19 of this appendix.

The NRC has an extended storage program evaluating extended cask storage for durations up to 300 years. Ongoing research into the extended storage of spent fuel is part of the NRC's effort to continuously evaluate and update its safety regulations. As noted in Appendix B, the NRC is not aware of any deficiencies in its current regulations that would challenge the determination that continued safe storage of spent fuel in dry casks is feasible.

The environmental impacts of geologic disposal are out-of-scope for this proceeding. Any geologic disposal casks would be evaluated and approved prior to the operation of a geologic disposal facility.

Recommended uses of the heat emitted from dry casks are also outside the scope of the Rule and GEIS. No changes were made to the GEIS or Rule as a result of these comments.

(3-3) (50-3) (63-7) (100-1) (163-14-3) (205-6) (230-12) (245-24-1) (245-13-6) (246-17-4) (249-8) (249-9) (262-2) (279-2) (280-7) (326-63-6) (327-13-2) (327-8-2) (328-8-3) (328-14-4) (328-14-5) (328-14-7) (329-20-3) (377-4-12) (377-4-15) (377-5-2) (377-5-5) (419-15) (552-3-3) (640-6) (698-3) (698-4) (698-5) (819-12) (883-4) (901-2) (910-11) (919-1-16) (919-5-16) (919-1-17) (919-1-8) (928-1) (929-10) (929-12) (929-13) (929-14) (929-19) (929-20) (929-4)

D.2.38.6 – COMMENT: Many commenters expressed concern over storage of spent fuel in pools. Commenters stated that the spent fuel pools were not intended for long-term storage of spent fuel over periods as long as decades. Rather, storage was intended only to last until the spent fuel had cooled for removal from the pool, and that it is irresponsible to allow continued pool storage. Commenters stated that the quantity of spent fuel stored in the pools exceeds the original design basis by up to 9 times and that the pools can contain up to 40 times more nuclear material than reactor cores; almost 80 percent of all spent fuel is still in the pools. Commenters stated that the pools are not protected by redundant emergency makeup and cooling systems and they lack robust containment structures. Commenters stated that overcrowding of spent fuel pools is a problem, which increases the potential for a radioactive release that could put the surrounding communities and the nation at risk of a potential catastrophic accident or terrorist event that could result in land contamination. Commenters stated that spent fuel pools are vulnerable to power outages, earthquakes, meltdowns, and terrorist attacks, which can lead to leaks and area contamination. Commenters stated that NRC cannot dismiss pool accidents as improbable or of low probability. Commenters indicated that while the dry casks at Fukushima survived, the spent fuel pools are collapsing, and thus the Fukushima event should be instructive to the United States and prompt removal of spent fuel from pools.

Commenters expressed support for transferring spent fuel to dry cask as a national priority. Several commenters cited NUREG–1738 (NRC 2001b), *Technical Study of Spent Fuel Pool Accident Risk at Decommissioning Nuclear Power Plants*, to support the view that dry cask storage of spent fuel is safer than pool storage.

Commenters stated that (1) reactors should not be allowed to generate additional spent fuel until all existing spent fuel has been removed from the pools, (2) facilities that have been shut down should be required to transfer all spent fuel to casks before gaining access to decommissioning funds, and (3) the NRC should require that any refueling event result in a net transfer of spent fuel to cask storage.

Some commenters indicated that pool storage is safe and pools have safely contained spent fuel for over 30 years, which is equivalent to over 3,000 years of operating experience with no significant environmental impact.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

August 19, 2015

Mr. Thomas J. Palmisano
Vice President and Chief Nuclear Officer
Southern California Edison Company
San Onofre Nuclear Generating Station
P.O. Box 128
San Clemente, CA 92674-0128

SUBJECT: SAN ONOFRE NUCLEAR GENERATING STATION, UNITS 2 AND 3 – REVIEW AND APPROVAL OF THE IRRADIATED FUEL MANAGEMENT PLAN (TAC NOS. MF4894 AND MF4895)

Dear Mr. Palmisano:

Pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR), Section 50.54(bb), licensees of nuclear power plants within 2 years following permanent cessation of operation must submit to the U.S. Nuclear Regulatory Commission (NRC), for review and preliminary approval, the program by which the licensee intends to manage and provide funding for the management of all irradiated fuel at the reactor, until title and possession of the fuel is transferred to the Secretary of Energy for its ultimate disposal in a repository. In addition, pursuant to Section 50.82(a)(4)(i), the licensee must submit a post-shutdown decommissioning activities report (PSDAR). A site-specific decommissioning cost estimate (DCE), containing the projected cost of managing irradiated fuel, is part of the PSDAR. On June 12, 2013, SCE informed the NRC that it had permanently ceased operations of SONGS Units 2 and 3 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML131640201).

By letter dated September 23, 2014 (ADAMS Accession No. ML14269A032), Southern California Edison Company (SCE, the licensee) submitted the San Onofre Nuclear Generating Station (SONGS), Units 2 and 3, Irradiated Fuel Management Plan (IFMP) to the NRC. SCE concurrently submitted the PSDAR and the site-specific DCE under separate cover letters (ADAMS Accession Nos. ML14269A033 and ML14269A034, respectively). As approved by exemption dated September 5, 2014, (ADAMS Accession No. ML14101A132), SCE uses the nuclear decommissioning trust fund (DTF) for license termination, irradiated fuel management and site restoration expenditures. While costs associated with all of these activities are discussed in the IFMP, the enclosed review focuses on irradiated fuel management. The NRC staff is conducting a separate review of the PSDAR and site-specific DCE.

Based on its review of SCE's submittal, the NRC staff finds that the licensee's program to manage and provide funding for the management of all irradiated fuel is adequate and provides sufficient detail regarding the associated funding mechanisms. Further, the staff has determined that the elected actions within the program are consistent with NRC requirements for licensed possession of irradiated nuclear fuel and that these actions will be implemented in a timely basis. Therefore, the staff concludes that the SONGS, Units 2 and 3, IFMP complies with 10 CFR 50.54(bb) and approves the plan on a preliminary basis. The NRC staff's review of the SONGS IFMP is enclosed.

The NRC staff recognizes that the IFMP analysis is based on a reported DTF balance that may fluctuate over time. Should a material decline in the DTF balance occur, the staff's analysis and findings may be impacted. However, in accordance with 10 CFR 50.82(a)(8)(vii), the licensee must annually submit to the NRC, by March 31, a report on the status of its funding for managing irradiated fuel. Further, in accordance with 10 CFR 50.54(bb), the licensee shall notify the NRC of any significant changes to the IFMP. Accordingly, the regulations provide a means of informing the NRC staff of fluctuations in the reported DTF balance and significant changes to the IFMP.

If you have any questions, please contact me at 301-415-4037 or Thomas.Wengert@nrc.gov.

Sincerely,

Thomas J. Wengert

Thomas J. Wengert, Senior Project Manager
Plant Licensing IV-2 and Decommissioning
Transition Branch
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-361 and 50-362

Enclosure:
Safety Evaluation

cc w/enclosure: Distribution via Listserv

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

IRRADIATED FUEL MANAGEMENT PLAN

SOUTHERN CALIFORNIA EDISON COMPANY

SAN ONOFRE NUCLEAR GENERATING STATION, UNITS 2 AND 3

DOCKET NUMBERS 50-361 AND 50-362

1.0 INTRODUCTION AND BACKGROUND

By letter dated September 23, 2014 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML14269A032), Southern California Edison Company (SCE, the licensee) submitted the San Onofre Nuclear Generating Station (SONGS), Units 2 and 3, Irradiated Fuel Management Plan (IFMP) to the U.S. Nuclear Regulatory Commission (NRC). SCE concurrently submitted the Post-Shutdown Decommissioning Activities Report (PSDAR) and the Site Specific Decommissioning Cost Estimate (DCE) by separate letters (ADAMS Accession Nos. ML14269A033 and ML14269A034, respectively), which are currently under staff review.

2.0 BACKGROUND

As described in the SONGS PSDAR, the SONGS site is located on the coast of Southern California in San Diego County, and is approximately 62 miles southeast of Los Angeles and 51 miles northwest of San Diego. The property on which the units were built is subject to an easement from the United States Navy. The site is located entirely within the boundaries of the United States Marine Corps Base Camp Pendleton. The property is approximately 4,500 feet long and 800 feet wide, and encompasses 84 acres. The property is situated between the coast of the Pacific Ocean and Interstate 5 (I-5), but does not include the office buildings and facilities located east of I-5. The nearest privately owned land is approximately 2.5 miles away.

SONGS is a two-unit pressurized-water reactor site that houses supporting facilities. The reactors were previously licensed to produce 3,438 megawatt thermal each. A third unit (SONGS, Unit 1) existed until its closure in 1992. An onsite Independent Spent Fuel Storage Installation (ISFSI), used to store fuel from Units 1, 2, and 3 is located on the portion of the site previously occupied by Unit 1. Fuel storage at the ISFSI was initiated in 2003, and the pad was expanded in 2007 to support 63 horizontal storage modules. To date, a total of 51 dry storage containers (DSCs) have been installed, with 50 containers storing irradiated fuel and one containing greater-than-Class-C (GTCC) waste.

Enclosure

SONGS, Units 2 and 3, have been owned by four entities. SCE is authorized to act as the agent for the other owners. The percent ownership of both reactors is as follows: SCE owns 78.21 percent; San Diego Gas & Electric Company owns 20 percent; and Riverside owns 1.79 percent, with Anaheim providing decommissioning funding, despite not currently owning any percentage of the facilities. The relative obligation for decommissioning varies by unit and entity as follows:

Cost Categories	Owners			
	SDG&E	Riverside	Anaheim	SCE
SONGS 1	20%	0%	0%	80%
SONGS 2	20%	1.79%	2.4737%	75.7363%
SONGS 3	20%	1.79%	2.4625%	75.7475%
Common Facilities (Units 2 & 3)	20%	1.79%	2.4681%	75.7419%
SONGS 1 Fuel	20%	0%	0%	80%
SONGS 2/3 Fuel	20%	1.79%	2.3398%	75.8702%
ISFSI Maintenance and D&D	20%	1.6066%	2.2686%	76.1248%
San Diego Switchyard	100%	0%	0%	0%
Edison Switchyard	0%	0%	0%	100%
Interconnection Facilities	50%	0%	0%	50%
Nuclear Fuel Cancellation Charges	20%	1.79%	0%	78.21%

By letter dated June 12, 2013, SCE notified the NRC of its permanent cessation of operations of Units 2 and 3, effective on June 7, 2013 (ADAMS Accession No. ML131640201). SCE subsequently submitted two letters to the NRC, dated July 22, 2013 (ADAMS Accession No. ML13204A304), and June 28, 2013 (ADAMS Accession No. ML13183A391), certifying the permanent removal of fuel from the reactor vessels of Units 2 and 3, respectively.

The NRC staff notes that as approved by exemption dated September 5, 2014, (ADAMS Accession No. ML14101A132), SCE uses the nuclear decommissioning trust fund (DTF) for license termination, irradiated fuel management and site restoration expenditures. While costs associated with all of these activities are discussed in the IFMP, this review focuses specifically on the costs associated with the management of irradiated fuel. A separate review of the PSDAR and site-specific DCE is currently being performed by the NRC staff.

3.0 REGULATORY EVALUATION

3.1 Regulatory Requirements

Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.54(bb) states, in part:

> For nuclear power reactors licensed by the NRC, the licensee shall, within 2 years following permanent cessation of operation ... submit written notification to the Commission for its review and preliminary approval of the program by which the licensee intends to manage and provide funding for the management of all irradiated fuel at the reactor following permanent cessation of the operation

> of the reactor until title to the irradiated fuel and possession of the fuel is transferred to the Secretary of Energy for its ultimate disposal in a repository.

Section 50.54(bb) of 10 CFR further states:

> The licensee must demonstrate to NRC that the elected actions will be consistent with NRC requirements for licensed possession of irradiated nuclear fuel and that the actions will be implemented on a timely basis. Where implementation of such actions requires NRC authorizations, the licensee shall verify in the notification that submittals for such actions have been or will be made to NRC and shall identify them. A copy of the notification shall be retained by the licensee as a record until expiration of the reactor operating license. The licensee shall notify the NRC of any significant changes in the proposed waste management program as described in the initial notification.

In addition, 10 CFR 50.82(a)(4)(i) states, in part, that the site-specific DCE that is submitted as part of the PSDAR includes the projected costs of managing irradiated fuel.

3.2 Information Submitted in Support of the IFMP Review

Similar to reviews of other IFMPs,[1] the NRC staff reviewed the following information submitted in support of the SONGS IFMP:

- Estimated cost to isolate the spent fuel pool (SFP) and fuel handling systems. For the decontamination (DECON) option, the cost to isolate the SFP and fuel handling systems may be considered as part of the preparation for DECON;
- Estimated cost to construct an ISFSI or a combination of wet/dry storage;
- Estimated annual cost for the operation of the selected option (wet or dry storage or a combination of the two) until the Department of Energy (DOE) takes possession of the fuel;
- Estimated cost for the preparation, packaging, and shipping of the fuel to DOE;
- Estimated cost to decommission the spent fuel storage facility; and
- Brief discussion of the selected storage method or methods, and the estimated time for these activities.

In addition, the NRC has determined that irradiated fuel can be safely stored in spent fuel pools and ISFSIs. The technical feasibility of either storage method was codified in the Continued Storage of Spent Nuclear Fuel Rule (79 FR 56238), as supported by NUREG-2157, "Generic Environmental Impact Statement for Continued Storage of Spent Nuclear Fuel" (ADAMS Accession No. ML14196A105), and specifically, Appendix B, "Technical Feasibility of Continued Storage and Repository Availability." With regard to "actions implemented on a timely basis," NUREG-2157 considers three time periods: short-term storage, long-term storage, and indefinite storage. While all storage timeframes are considered technically feasible, the short-term storage period of 60 years beyond licensed life for reactor operations covers the IFMP

[1] Most recently, the safety evaluation by the Office of Nuclear Reactor Regulation related to the updated IFMP of Duke Energy Florida, Inc., Crystal River Unit 3 Nuclear Generating Plant, Docket No. 50-302 (ADAMS Accession No. ML14344A408).

proposed by SCE. This timeframe coincides with the decommissioning timeframe. A minimum assumption is that all spent fuel will be moved from the spent fuel pool to dry cask storage by the end of the short-term storage timeframe.

4.0 TECHNICAL EVALUATION

The SONGS IFMP represents a high level plan for the management of irradiated fuel. It references the SONGS DCE as identifying the details, schedules, and costs of the spent fuel management activities. As noted above, the NRC is reviewing the SONGS DCE and PSDAR separately. However, during this review, the NRC staff considered relevant portions of the DCE and ensured consistency between the documents.

Table 1 of the IFMP identifies the seven periods of spent fuel management. For each period, the table provides a brief description, the duration, and the cost on a unit basis in 2014 dollars in the unit of thousands. The first period, "Spent Fuel Management Transition," consists of activities that support the implementation of security enhancements required for reductions in staff, cyber security modifications, post-Fukushima modifications for Unit 2, and the design and fabrication of spent fuel canisters. This period began in June 2013, ended in December 2013, and cost a total of $129,997,000. As per the IFMP, the safe initial interim storage of SONGS irradiated fuel will occur in each unit's respective SFP (also known as "wet storage"). The normal systems that support the SFPs will be replaced by stand-alone cooling and filtration systems. These new systems will allow the SFP to independently operate from the normal systems (also known as "islanding"). Table 2 of the IFMP provides the estimated cost to isolate the SFPs and fuel handling systems, which is $22,183,000. After appropriate cooling has occurred, all irradiated fuel in the SFPs will be transferred to the ISFSI for "dry storage." This activity is currently scheduled to be completed by 2019.

The second period, "Spent Fuel Transfer to Dry Storage," includes preparation and issuance of the IFMP; selection of the dry storage system canister design and vendor; design and construction of the ISFSI expansion (as discussed below); purchase, delivery, and loading of spent fuel canisters; and the transfer of the fuel to the ISFSI. This period began in January 2014 and is expected to end in June 2019. It is estimated to cost $716,822,000.

Units 2 and 3 have generated a total of 3,460 irradiated fuel assemblies. At present, 792 irradiated fuel assemblies from both units have already been transferred to the ISFSI. The remaining 2,668 irradiated fuel assembles will be loaded into DSCs and transferred to the ISFSI. The ISFSI currently contains 18 DSCs that store Unit 1 fuel and 33 DSCs that store Units 2 and 3 fuel. All of the fuel that is currently stored on the ISFSI is kept in Transnuclear NUHOMS Model Number-24PT1 or PT4 DSCs.

SCE intends to expand the current ISFSI in order to accommodate the remaining irradiated fuel from Units 2 and 3. Additional DSCs will be procured from one or more of the available, NRC-approved dry storage system suppliers, which began in 2014. An estimated 47 DSCs will be required for Unit 2 fuel, and an estimated 44 DSCs will be required for Unit 3 fuel. The exact number will depend on the capacity of the selected system and the number of DSCs needed to store GTCC waste and other materials. The estimated cost for a combination of wet/dry storage and ISFSI expansion is $306,391,000.

The third period, "Dry Storage during Decommissioning for Units 1, 2, and 3," is scheduled for June 2019 through December 2031. The execution of scheduled activities during this period is expected to cost a total of $122,849,000. The fourth period, "Dry Storage Only – Units 1, 2, and 3," is scheduled for December 2031 through December 2049 and is expected to cost $58,765,000. The fifth period, "Dry Storage Only – Units 2 and 3," is scheduled for December 2049 through September 2051, and is expected to cost $214,653,000.

The sixth period, "Decontamination and Dismantlement (D&D) Period 1," is scheduled for December 2049 through May 2050 and is expected to cost $2,520,000. The final period, "D&D Period 2," is scheduled for May 2050 through September 2051 and is expected to cost $30,590,000. These final two periods will serve as the time to decontaminate and dismantle the ISFSI and return the area to unrestricted use, once all spent fuel has been removed from the site.

The SONGS Units 2 and 3 IFMP is based on the commencement of industry-wide acceptance of spent fuel by DOE in 2024 and SONGS' priority-ranking in that queue. As such, SCE is assuming that all fuel will be removed from the SONGS site by 2049. The estimated cost for preparation, packing, and shipping of the fuel to DOE is $6,742,000. The estimated cost to decommission the ISFSI is $33,110,000.

The NRC staff, as part of its analysis of the IFMP, used the information and cost estimates outlined above, in conjunction with Tables 4A and 4B of the SONGS IFMP that provides the annual cost to manage the spent fuel, to calculate the ending balance for the SONGS DTF at the end of the projected fuel removal period. The calculation resulted in a positive ending balance: $406,084,000 for Unit 2 and $499,465,000 for Unit 3. The NRC staff subtracted projected radiological decontamination costs, spent fuel management costs, and site restoration costs from the projected opening balance on a yearly basis. The NRC staff then applied a 2-percent real rate of return on this value to calculate a projected year-end balance. The yearly closing balance calculations can be found in Attachment 1, "Unit 2 IFMP Closing Balance Calculations," and Attachment 2, "Unit 3 IFMP Closing Balance Calculations," of SCE's IFMP submittal.

The NRC staff finds the SONGS IFMP estimates to be reasonable, based on a cost comparison with similar decommissioning reactors, while acknowledging that there are large uncertainties and potential site-specific variances that may impact these cost estimates in the future.

Regarding the provision in 10 CFR 50.54(bb), "The licensee must demonstrate to NRC that the elected actions will be consistent with NRC requirements for licensed possession of irradiated nuclear fuel and that the actions will be implemented on a timely basis," the SONGS IFMP is consistent with the determinations that the NRC has made in the Continued Storage of Spent Nuclear Fuel Rule and NUREG-2157. The NRC staff has determined that storing fuel in either the spent fuel pool or ISFSI represents an acceptable means for storing irradiated fuel. The licensee's plan contains both storage methods, with irradiated fuel being taken out of the spent fuel pool and fully transitioned to the ISFSI within 5 years, followed by complete dry storage. The anticipated date to transfer fuel to DOE and subsequent decommissioning of the ISFSIs are scheduled to be completed in 2051. This supports the requirement to complete decommissioning within the 60-year timeframe, as required by 10 CFR 50.82.

5.0 CONCLUSION

Based on the NRC staff's review of the SONGS IFMP and site-specific DCE, the staff finds that SCE has provided sufficient detail to satisfy the requirements of 10 CFR 50.54(bb). Based on the staff's calculated positive ending balance (as provided in Attachments 1 and 2 of this safety evaluation), the NRC staff finds that SCE has demonstrated reasonable assurance that funding will be available to maintain the IFMP until the fuel is transferred to the DOE for permanent disposal. Further, the NRC staff finds that the actions and timeframes described in the IFMP are consistent with the NRC's generic determination for spent fuel management, associated with the Continued Storage of Spent Nuclear Fuel Rule, as supported by NUREG-2157. Therefore, the NRC staff preliminarily approves the SONGS IFMP.

Principal Contributor: Eric Olvera

Date: August 19, 2015

SONGS Unit 2: IFMP Closing Balance Calculations

Year	Opening Balance	Radiological Decontamination	Spent Fuel Management	Site Restoration	2% Interest	Closing Balance
2013						$1,847,000
2014	$1,847,000	$79,799	$35,719	$15,089	$34,328	$1,750,721
2015	$1,750,721	$69,196	$106,308	$7,439	$31,356	$1,599,133
2016	$1,599,133	$54,541	$59,308	$3,730	$29,631	$1,511,186
2017	$1,511,186	$111,903	$59,308	$1,957	$26,760	$1,364,778
2018	$1,364,778	$47,520	$59,308	$0	$25,159	$1,283,109
2019	$1,283,109	$108,328	$27,554	$13,539	$22,674	$1,156,362
2020	$1,156,362	$185,482	$4,908	$36	$19,319	$985,254
2021	$985,254	$79,081	$4,908	$36	$18,025	$919,254
2022	$919,254	$54,785	$4,908	$1,927	$17,153	$874,787
2023	$874,787	$158,207	$4,908	$36	$14,233	$725,868
2024	$725,868	$37,930	$4,908	$16,848	$13,324	$679,506
2025	$679,506	$2,922	$4,908	$44,621	$12,541	$639,596
2026	$639,596	$2,922	$4,908	$19,412	$12,247	$624,601
2027	$624,601	$2,922	$4,908	$22,469	$11,886	$606,188
2028	$606,188	$2,922	$4,908	$31,688	$11,333	$578,004
2029	$578,004	$2,922	$4,908	$66,873	$10,066	$513,367
2030	$513,367	$2,922	$4,908	$71,867	$8,673	$442,343
2031	$442,343	$2,055	$5,089	$23,181	$8,240	$420,258
2032	$420,258	$2,122	$7,214	$0	$8,218	$419,141
2033	$419,141	$0	$7,214	$0	$8,239	$420,165
2034	$420,165	$0	$7,214	$0	$8,259	$421,210
2035	$421,210	$0	$7,228	$0	$8,280	$422,262
2036	$422,262	$0	$7,665	$0	$8,292	$422,889
2037	$422,889	$0	$7,665	$0	$8,304	$423,528
2038	$423,528	$0	$7,665	$0	$8,317	$424,181
2039	$424,181	$0	$7,665	$0	$8,330	$424,846
2040	$424,846	$0	$7,665	$0	$8,344	$425,525
2041	$425,525	$0	$7,665	$0	$8,357	$426,217
2042	$426,217	$0	$7,665	$0	$8,371	$426,923
2043	$426,923	$0	$7,665	$0	$8,385	$427,643
2044	$427,643	$0	$7,665	$0	$8,400	$428,378
2045	$428,378	$0	$7,665	$0	$8,414	$429,127
2046	$429,127	$0	$7,665	$0	$8,429	$429,891
2047	$429,891	$0	$7,665	$0	$8,445	$430,671
2048	$430,671	$0	$7,665	$0	$8,460	$431,466
2049	$431,466	$0	$7,667	$0	$8,476	$432,275
2050	$432,275	$0	$9,974	$20,177	$8,042	$410,166
2051	$410,166	$0	$6,573	$11,928	$7,833	$399,498
2052	$399,498	$0	$0	$1,377	$7,962	$406,084
Totals		$1,008,481	$559,311	$374,230		

Notes (SONGS IFMP):
Costs are in 2014 dollars (in thousands) and are not escalated from the base year.
SONGS Unit 2 Trust fund balances at end of 2013 were $1,847,000.
Radiological Decontamination, Spent Fuel Management, and Site Restoration figures from SONGS IFMP.

Attachment 1

SONGS Unit 3: IFMP Closing Balance Calculations						
Year	Opening Balance	Radiological Decontamination	Spent Fuel Management	Site Restoration	2% Interest	Closing Balance
2013						$2,079,400
2014	$2,079,400	$78,964	$40,156	$15,969	$38,886	$1,983,197
2015	$1,983,197	$74,096	$112,024	$9,390	$35,754	$1,823,441
2016	$1,823,441	$61,451	$64,405	$25,227	$33,447	$1,705,805
2017	$1,705,805	$40,631	$64,405	$3,799	$31,939	$1,628,910
2018	$1,628,910	$86,348	$64,405	$0	$29,563	$1,507,720
2019	$1,507,720	$96,521	$29,675	$13,908	$27,352	$1,394,968
2020	$1,394,968	$120,873	$4,908	$2,135	$25,341	$1,292,393
2021	$1,292,393	$194,090	$4,908	$575	$21,856	$1,114,676
2022	$1,114,676	$135,313	$4,908	$2,467	$19,440	$991,428
2023	$991,428	$114,581	$4,908	$1,511	$17,409	$887,837
2024	$887,837	$26,874	$4,908	$36,778	$16,386	$835,662
2025	$835,662	$2,922	$4,908	$40,655	$15,744	$802,921
2026	$802,921	$2,922	$4,908	$21,676	$15,468	$788,883
2027	$788,883	$2,922	$4,908	$25,848	$15,104	$770,309
2028	$770,309	$2,922	$4,908	$20,945	$14,831	$756,365
2029	$756,365	$2,922	$4,908	$117,321	$12,624	$643,838
2030	$643,838	$2,922	$4,908	$116,672	$10,387	$529,723
2031	$529,723	$2,055	$5,089	$25,501	$9,942	$507,019
2032	$507,019	$2,122	$7,214	$0	$9,954	$507,637
2033	$507,637	$0	$7,214	$0	$10,008	$510,432
2034	$510,432	$0	$7,214	$0	$10,064	$513,282
2035	$513,282	$0	$7,228	$0	$10,121	$516,175
2036	$516,175	$0	$7,665	$0	$10,170	$518,680
2037	$518,680	$0	$7,665	$0	$10,220	$521,236
2038	$521,236	$0	$7,665	$0	$10,271	$523,842
2039	$523,842	$0	$7,665	$0	$10,324	$526,500
2040	$526,500	$0	$7,665	$0	$10,377	$529,212
2041	$529,212	$0	$7,665	$0	$10,431	$531,978
2042	$531,978	$0	$7,665	$0	$10,486	$534,799
2043	$534,799	$0	$7,665	$0	$10,543	$537,677
2044	$537,677	$0	$7,665	$0	$10,600	$540,612
2045	$540,612	$0	$7,665	$0	$10,659	$543,606
2046	$543,606	$0	$7,665	$0	$10,719	$546,660
2047	$546,660	$0	$7,665	$0	$10,780	$549,775
2048	$549,775	$0	$7,665	$0	$10,842	$552,952
2049	$552,952	$0	$7,667	$0	$10,906	$556,191
2050	$556,191	$0	$9,974	$23,120	$10,462	$533,559
2051	$533,559	$0	$6,573	$45,566	$9,628	$491,048
2052	$491,048	$0	$0	$1,377	$9,793	$499,465
Totals		$1,051,451	$586,876	$550,440		

Notes (SONGS IFMP):
Costs are in 2014 dollars (in thousands) and are not escalated from the base year.
SONGS Unit 3 Trust fund balances at end of 2013 were $2,079,400.
Radiological Decontamination, Spent Fuel Management, and Site Restoration figures from SONGS IFMP.

Attachment 2

The NRC staff recognizes that the IFMP analysis is based on a reported DTF balance that may fluctuate over time. Should a material decline in the DTF balance occur, the staff's analysis and findings may be impacted. However, in accordance with 10 CFR 50.82(a)(8)(vii), the licensee must annually submit to the NRC, by March 31, a report on the status of its funding for managing irradiated fuel. Further, in accordance with 10 CFR 50.54(bb), the licensee shall notify the NRC of any significant changes to the IFMP. Accordingly, the regulations provide a means of informing the NRC staff of fluctuations in the reported DTF balance and significant changes to the IFMP.

If you have any questions, please contact me at 301-415-4037 or Thomas.Wengert@nrc.gov.

Sincerely,

/RA/

Thomas J. Wengert, Senior Project Manager
Plant Licensing IV-2 and Decommissioning
Transition Branch
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-361 and 50-362

Enclosure:
Safety Evaluation

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NAME	SKoenick	PBlechman	ABowers	MSampson
DATE	7/9/15	7/9/15	7/10/15	7/17/15
OFFICE	OGC	NRR/DORL/LPL4-2/BC	NRR/DORL/D(A)	NRR/DORL/LPL4-2/PM
NAME	BMizuno w/cmt	MKhanna	ALLund/GWilson for	TWengert
DATE	8/18/15	8/10/15	8/12/15	8/19/15

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

August 20, 2015

Mr. Thomas J. Palmisano
Vice President and Chief Nuclear Officer
Southern California Edison Company
San Onofre Nuclear Generating Station
P.O. Box 128
San Clemente, CA 92674-0128

SUBJECT: SAN ONOFRE NUCLEAR GENERATING STATION UNITS 2 AND 3 – REVIEW OF POST-SHUTDOWN DECOMMISSIONING ACTIVITIES REPORT (TAC NOS. MF4892 AND MF4893)

Dear Mr. Palmisano:

By letter dated June 12, 2013 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML131640201), Southern California Edison (SCE or the licensee) submitted a certification to the U.S. Nuclear Regulatory Commission (NRC) indicating that, effective June 7, 2013, SCE had permanently ceased power operations at Units 2 and 3 of the San Onofre Nuclear Generating Station (SONGS). By letters dated June 28, and July 22, 2013 (ADAMS Accession Nos. ML13183A391 and ML13204A304), SCE certified that it had permanently defueled the SONGS Unit 3 and Unit 2 reactor vessels, respectively. As permanently shutdown and defueled facilities, and in accordance with Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, Section 50.82(a)(2), SCE is no longer authorized to operate the reactor or emplace nuclear fuel into the SONGS Units 2 or 3 reactor vessels. SONGS Units 2 and 3 are still authorized to possess and store irradiated nuclear fuel. Irradiated fuel is currently being stored onsite in spent fuel pools (SFPs) and in the Independent Spent Fuel Storage Installation (ISFSI) dry casks.

On September 23, 2014, SCE submitted to the NRC the Post-Shutdown Decommissioning Activities Report (PSDAR) and the Site-Specific Decommissioning Cost Estimate (DCE) (ADAMS Accession No. ML14272A121) for SONGS, Units 2 and 3, pursuant to 10 CFR Part 50.82(a)(4). The public receipt of the original PSDAR was noticed in the *Federal Register* on October 14, 2014 (79 FR 61668). The purpose of this letter is to provide you with the results of the NRC staff's review of the PSDAR.

The purposes of the PSDAR and DCE are to: (1) inform the public of the licensee's planned decommissioning activities, (2) assist in the scheduling of NRC resources necessary for the appropriate oversight activities, (3) ensure that the licensee has considered all of the costs of the planned decommissioning activities and has considered the funding for the decommissioning process, and (4) ensure that the environmental impacts of the planned decommissioning activities are bounded by those considered in existing environmental impact statements.

Pursuant to 10 CFR Part 50.82(a)(4)(i), the PSDAR must contain a description of the planned decommissioning activities along with a schedule for their accomplishment, a discussion that provides the reasons for concluding that the environmental impacts associated with site-specific

decommissioning activities will be bounded by appropriate previously issued environmental impact statements, and a site-specific DCE, including the projected cost of managing irradiated fuel. SCE's irradiated fuel management plan (IFMP) was submitted to the NRC on September 23, 2014 (ADAMS Accession No. ML14269A032). While the NRC staff considered public comments about the projected costs of managing irradiated fuel within the PSDAR review, the staff conducted a separate evaluation of the IFMP, as required in 10 CFR Part 50.54(bb). The results of the staff's review of the IFMP is documented in an NRC safety evaluation dated August 19, 2015 (ADAMS Accession No. ML15182A256).

The NRC staff held a public meeting in the vicinity of SONGS, Units 2 and 3, on October 27, 2014, to describe the decommissioning process, receive comments, and answer questions regarding the PSDAR. A summary of the meeting, dated March 9, 2015, can be found at ADAMS Accession No. ML14351A396. Public questions and comments on the PSDAR and other areas related to the site's decommissioning, including the NRC staff's responses, are available for review in the transcript of the meeting (ADAMS Accession No. ML14336A252). The NRC staff also received comments electronically from the public concerning the PSDAR. Public comments submitted electronically can be viewed at http://www.regulations.gov, by searching on Docket NRC-2014-0223 and selecting "Open Docket Folder."

Public comments from the PSDAR public meeting and those submitted electronically to the NRC, generally fell into two categories: (1) questions and comments that are within the regulatory purview of the NRC staff's review of the PSDAR, and considered by the staff during its review, and (2) questions and comments that, upon review, were found to be outside the regulatory authority of the NRC, or were not relevant to the review performed by the NRC staff (i.e., whether the licensee's PSDAR meets the requirements of 10 CFR 50.82(a)(4)) and, thus, were not considered.

The public questions and comments that the NRC staff considered during its review of the PSDAR are summarized below. Details of the specific questions or comments can be found in the documents referenced above.

- Questions or comments about whether there is reasonable assurance that sufficient funds are available to decommission the facility, who manages those funds, and the NRC's role in oversight and monitoring the use of these funds.

- Questions or comments about the specific planned activities the licensee listed regarding the decommissioning of the facility. This includes questions about the removal of facility structures. Also considered were comments on the decommissioning process in general, including the role of other governmental agencies that are participating in the decommissioning process (i.e., other than the NRC), and future opportunities for public involvement.

- Questions or comments about the date that the permanent national high level waste storage facility is assumed to be available.

- Questions or comments about the planned availability of the Southern California Edison emergency response center in the City Hall during decommissioning.

- Questions or comments about the lack of specificity in the decommissioning plan, including recordkeeping, and detailed cost estimates.
- Questions or comments about emergency preparedness.
- Questions or comments about the environmental impact of removing the intake and discharge canals.
- Questions or comments related to the projected cost of managing irradiated fuel.

Public comments or questions that, upon review, were found to be outside of the NRC's regulatory purview or outside the scope of the NRC staff's review of the PSDAR, as defined in 10 CFR 50.82(a)(4)(i), are summarized below.

- Questions or comments about the appropriateness or reasonableness of allowing the licensee to pass maintenance and decommissioning costs on to the customer.
- Questions or comments about security regulations and the design-basis threat.
- Questions or comments about the use of the facility after decommissioning is completed.
- Questions or comments about the appropriateness of applying generic evaluations published by the NRC to specific facilities, such as SONGS.
- Questions or comments about the national policy on long term storage of nuclear fuel, except for assumptions regarding the availability of a permanent national high level waste storage facility.
- Questions or comments about the NRC's regulatory authority during the decommissioning period.
- Questions or comments about the adequacy and acceptability of current NRC regulations.
- Questions or comments about federal laws currently undergoing promulgation.
- Questions or comments about the acceptable radiation exposure limit requirements.
- Questions or comments about the performance, design requirements, and the availability of inspection and repair methods of spent fuel storage casks previously certified or under review by the NRC (Note: these issues are addressed during NRC's licensing of the spent fuel storage casks, or during cask license renewal).
- Questions or comments about the licensee's choice of decommissioning method.
- Requests to initiate research are outside the scope of this review and were not considered.

The NRC staff reviewed the PSDAR and DCE against the requirements in 10 CFR 50.82(a). In addition, the staff used the guidance in Regulatory Guide (RG) 1.185, Revision 1 (RG 1.185), "Standard Format and Content for Post-Shutdown Decommissioning Activities Report," dated June 2013 (ADAMS Accession No. ML13140A038), in conducting its review and concludes the following:

1. Section II of the PSDAR, "Description of Planned Decommissioning Activities," and the DCE provide the applicable information identified in Section C(1) of RG 1.185. The NRC staff's review found that the licensee described the activities associated with the major periods or milestones related to the decommissioning, as required by 10 CFR 50.82(a)(4)(i) and consistent with RG 1.185. These periods included "Transition to Decommissioning," "Decommissioning Planning and Site Modifications," "Decommissioning Preps/Reactor Internals Segmentation," "Plant Systems and Large Component Removal," "Building Decontamination," and "License Termination During Demolition."

2. Section II of the PSDAR also provides the estimated dates for initiation and completion of major decommissioning activities, as required by 10 CFR 50.82(a)(4)(i), and consistent with Section C(2) of RG 1.185. The NRC staff finds that the schedule for decommissioning activities is adequate to achieve SONGS, Units 2 and 3, license termination within 60 years of permanent cessation of operations, as required by 10 CFR 50.82(a)(3). Additional schedule information regarding the major decommissioning activities, spent fuel management and site restoration was provided in Appendix C, "Detailed Project Schedule," of the DCE.

3. Section III of the PSDAR, "Estimate of Expected Decommissioning and Spent Fuel Management Costs," provides an estimate of the expected decommissioning costs for SONGS, Units 2 and 3. Section III of the PSDAR references the DCE for specifics regarding the estimated costs associated with decommissioning and spent fuel management. SCE estimated the total decommissioning cost of SONGS, Units 2 and 3 (license termination, spent fuel management, and site restoration), to be approximately $4.411 billion (in 2014 dollars). SCE estimated the costs associated with only the long-term irradiated fuel management to be $1.276 billion (in 2014 dollars). The NRC staff reviewed the cost estimates against the guidance in RG 1.185, Section C.3 and finds that SCE's site-specific DCE and the cost of long-term storage of spent fuel for SONGS, Units 2 and 3, are considered reasonable, are described consistent with the guidance in RG 1.185, provide sufficient details associated with the funding mechanisms, and meet the requirements of 10 CFR 50.82(a)(4)(i).

4. Section IV of the PSDAR, "Environmental Impacts," provides a discussion of the potential environmental impacts associated with the SONGS, Units 2 and 3, decommissioning activities, as identified by Section C(4) of RG 1.185. The PSDAR includes a comparison of potential environmental impacts from SONGS, Units 2 and 3, planned decommissioning activities with impacts from similar activities provided in NUREG-0586, Initial Report, "Final Generic Environmental Impact Statement [GEIS] on Decommissioning of Nuclear Facilities," dated August 1988 and Supplement 1, dated November 2002. The GEIS and supplement evaluated the environmental impacts of decommissioning activities at nuclear power reactors necessary to reduce the residual

radioactivity to levels that allow for the termination of the NRC license. The licensee compared the SONGS, Units 2 and 3, facility to the reference facility in NUREG-0586 and found that the SONGS, Units 2 and 3, environmental impacts were bounded by the analysis provided in NUREG-0586. After reviewing the licensee's comparison, the NRC staff finds that the potential environmental impacts associated with SONGS, Units 2 and 3, decommissioning activities are bounded by the previously issued GEIS and its supplements, are described consistent with the guidance in RG 1.185, and meet the requirements of 10 CFR 50.82(a)(4)(i).

Based on this review, the NRC staff finds that the PSDAR contains the information required by 10 CFR 50.82(a)(4)(i), and is consistent with RG 1.185. In accordance with 10 CFR 50.82(a)(7), SCE must notify the NRC in writing before performing any significant decommissioning activity inconsistent with, or making a significant schedule change from, the planned decommissioning activities or schedules described in the PSDAR, including changes that significantly increase the decommissioning costs.

In accordance with 10 CFR Part 2, "Agency Rules of Practice and Procedure," a copy of this letter will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records component of NRC's ADAMS. ADAMS is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html.

If you have any questions regarding this letter, please contact Thomas J. Wengert, at 301-415-4037 or by e-mail at Thomas.Wengert@nrc.gov.

Sincerely,

Christopher Gratton for

Thomas J. Wengert, Senior Project Manager
Plant Licensing IV-2 and Decommissioning
Transition Branch
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-361 and 50-362

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radioactivity to levels that allow for the termination of the NRC license. The licensee compared the SONGS, Units 2 and 3, facility to the reference facility in NUREG-0586 and found that the SONGS, Units 2 and 3, environmental impacts were bounded by the analysis provided in NUREG-0586. After reviewing the licensee's comparison, the NRC staff finds that the potential environmental impacts associated with SONGS, Units 2 and 3, decommissioning activities are bounded by the previously issued GEIS and its supplements, are described consistent with the guidance in RG 1.185, and meet the requirements of 10 CFR 50.82(a)(4)(i).

Based on this review, the NRC staff finds that the PSDAR contains the information required by 10 CFR 50.82(a)(4)(i), and is consistent with RG 1.185. In accordance with 10 CFR 50.82(a)(7), SCE must notify the NRC in writing before performing any significant decommissioning activity inconsistent with, or making a significant schedule change from, the planned decommissioning activities or schedules described in the PSDAR, including changes that significantly increase the decommissioning costs.

In accordance with 10 CFR Part 2, "Agency Rules of Practice and Procedure," a copy of this letter will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records component of NRC's ADAMS. ADAMS is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html.

If you have any questions regarding this letter, please contact Thomas J. Wengert, at 301-415-4037 or by e-mail at Thomas.Wengert@nrc.gov.

Sincerely,

/RA Christopher Gratton for

Thomas J. Wengert, Senior Project Manager
Plant Licensing IV-2 and Decommissioning
Transition Branch
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-361 and 50-362

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NAME	CGratton	TWengert	PBlechman	BWatson*
DATE	8/20/15	8/19/2015	8/13/15	8/13/15
OFFICE	NRR/DLR/RERB/BC	NRR/DIRS/IFIB/BC	NRR/DORL/LPL4-2/BC	NRR/DORL/LPL4-2/PM
NAME	DWrona*	TBowers**	MKhanna	CGratton
	8/19/15	4/22/2015	8/20/15	8/20/15

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POLICY ISSUE
(Information)

December 31, 2019

SECY-20-0001

FOR: The Commissioners

FROM: John W. Lubinski, Director
Office of Nuclear Material Safety
and Safeguards

SUBJECT: SUMMARY OF STAFF REVIEW AND FINDINGS OF THE 2019 DECOMMISSIONING FUNDING STATUS REPORTS FROM OPERATING AND DECOMMISSIONING POWER REACTOR LICENSEES

PURPOSE:

The purpose of this paper is to inform the Commission of the U.S. Nuclear Regulatory Commission (NRC) staff's findings from its review of the 2019 decommissioning funding status (DFS) reports submitted by operating power reactor licensees and power reactor licensees in decommissioning. This paper does not address any new commitments or resource implications.

BACKGROUND:

In 1988, the NRC established technical and financial requirements to assure that decommissioning of all licensed facilities would be accomplished in a safe and timely manner and that adequate licensee funds would be available for this purpose (Volume 53 of the *Federal Register* (FR), page 24018 (53 FR 24018); June 27, 1988). "Decommission," in accordance with Title 10 of the *Code of Federal Regulations* (10 CFR) 50.2, "Definitions," means to remove a facility or site safely from service and reduce residual radioactivity to a level that permits: (1) release of the property for unrestricted use and termination of the license; or (2) release of

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(301) 415-1309

the property under restricted conditions and termination of the license. Therefore, decommissioning, as used in NRC regulations, refers exclusively to radiological decommissioning.

In 1998, in response to the anticipated deregulation of the power generating industry, the NRC amended the decommissioning financial assurance rules under 10 CFR 50.75, "Reporting and recordkeeping for decommissioning planning," resulting in additional methods and flexibility for reactor licensees to provide financial assurance for decommissioning (63 FR 50465; September 22, 1998). Additionally, the amended regulations established the requirements that power reactor licensees report, on a biennial basis, the status of their decommissioning funds and on material changes to their external trust agreements and other financial assurance mechanisms.

In 2011, the NRC further amended its regulations to improve decommissioning planning and to reduce the likelihood that any current operating facility would become a legacy site[1] (76 FR 35512; June 17, 2011). As a result, under 10 CFR 50.82, "Termination of license," power reactor licensees in decommissioning are required to provide annual DFS reports to the NRC that include, among other things, information on decommissioning expenditures made during the previous calendar year, the remaining balance of decommissioning funds, and an estimate of the cost to complete decommissioning.

DISCUSSION:

Pursuant to NRC regulations at 10 CFR 50.75(f)(1) (for operating power reactors) and 10 CFR 50.82(a)(8)(v)–(vi) (for power reactors in decommissioning), licensees are required to submit DFS reports to the NRC. DFS reports are required every 2 years from operating power reactor licensees, annually from operating power reactor licensees that are within 5 years of the projected end of their operation or involved in a merger or acquisition, and annually from power reactor licensees in decommissioning. Licensees must submit these reports to the NRC by March 31 of the reporting year. The reports must provide specified information that will allow the agency to monitor the status of decommissioning funds for all power reactor licensees from the time they begin operating until their license is terminated.

For operating reactors, in accordance with 10 CFR 50.75(f)(1), the DFS reports must include: (1) the amount of decommissioning funds estimated to be required pursuant to 10 CFR 50.75(b) and 10 CFR 50.75(c); (2) the amount of decommissioning funds accumulated to the end of the calendar year preceding the date of the report; (3) a schedule of the annual amounts remaining to be collected; (4) the assumptions used regarding rates of escalation in decommissioning costs, rates of earnings on decommissioning funds, and rates of other factors used in funding projections; (5) any contracts on which the licensee is relying; (6) any modifications occurring to a licensee's current method of providing financial assurance since the last submitted report; and (7) any material changes to trust agreements.

10 CFR 50.75(c) requires licensees to demonstrate reasonable assurance of funding for decommissioning. Shortfalls should, therefore, be corrected in a timely manner. The staff notes that while the decommissioning funding amounts certified by licensees under this part do not represent the actual cost of plant decommissioning, they do provide assurance that licensees

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1 As defined in the Statement of Considerations accompanying the 2011 rule, a "legacy site" is a facility that is in decommissioning status with complex issues and an owner who cannot complete the decommissioning work for technical or financial reasons.

have available the bulk of the funds to safely decommission the facility. Adjustments to the certification amount are required annually over the operating life of the facility and account for inflation in the labor, energy, and waste burial components of decommissioning costs. Within 5 years before the projected end of operations, 10 CFR 50.75(f) requires that each licensee submit a preliminary decommissioning cost estimate that includes an updated assessment of the major factors that could affect the cost to decommission. The preliminary cost estimate is a more accurate representation of the licensee's cost to decommission as compared to the NRC required minimum. Therefore, shortfalls identified during the operating cycle and between biennial DFS reporting periods are considered to be temporary lapses in funding for decommissioning that may be remedied by use of a parent company guarantee, trust fund growth, or trust fund contributions. In any event, guidance in Regulatory Guide (RG) 1.159, "Assuring the Availability of Funds for Decommissioning Nuclear Reactors," Revision 2, issued October 2011, states that shortfalls identified in a biennial DFS report must be corrected by the time the next report is due.

For power reactors in decommissioning, in accordance with 10 CFR 50.82(a)(8)(v), the annual DFS reports must include: (1) the amount spent on decommissioning, both cumulative and over the previous calendar year, the remaining balance of any decommissioning funds, and the amount provided by other financial assurance methods being relied upon; (2) an estimate of the costs to complete decommissioning, reflecting any difference between actual and estimated costs for work performed during the year, and the decommissioning criteria upon which the estimate is based; (3) any modifications occurring to a licensee's current method of providing financial assurance since the last submitted report; and (4) any material changes to trust agreements or financial assurance contracts. Pursuant to 10 CFR 50.82(a)(8)(vi), if the sum of the balance of any remaining decommissioning funds, plus earnings on such funds calculated at not greater than a 2 percent real rate of return, together with the amount provided by other financial assurance methods being relied upon, does not cover the estimated cost to complete the decommissioning, the DFS report must include additional financial assurance to cover the estimated cost of completion.

Pursuant to 10 CFR 50.75(e)(2), the NRC reserves the right to review, as needed, the rate of accumulation of decommissioning funds and take additional actions as appropriate, on a case-by-case basis, to ensure a licensee's adequate accumulation of decommissioning funds. This includes modification of a licensee's schedule for the accumulation of decommissioning funds. Additionally, in accordance with 10 CFR 50.82(c), for licensees that shut down their reactors prematurely, the collection period for any shortfall of funds will be determined on a case-by-case basis upon application by the licensee, taking into account the specific financial situation of each licensee.

Using staff guidance in Office of Nuclear Reactor Regulation Office Instruction LIC-205, "Procedures for NRC's Independent Analysis of Decommissioning Funding Assurance for Operating Nuclear Power Reactors and Power Reactors in Decommissioning," Revision 6, dated April 10, 2017,[2] the NRC staff reviewed the 2019[3] DFS reports for completeness and compliance with 10 CFR 50.75(f)(1) - (2) and 10 CFR 50.82(a)(8)(v) - (vi). The staff's review included reports for 98 operating power reactors and 21 power reactors in decommissioning. Two tables summarizing the staff's review are enclosed. Table 1, "2019 Decommissioning Funding Status Report for Operating Power Reactor Licensees (December 31, 2018)," summarizes the information from the 98 DFS reports submitted by operating power reactor

2 Agencywide Documents Access and Management System (ADAMS) Accession No. ML17075A095
3 The 2019 DFS reports reflect the financial status as of December 31, 2018.

licensees,[4] and Table 2, "2019 Decommissioning Funding Status Report for Power Reactor Licensees in Decommissioning (December 31, 2018)," summarizes the information from the 21 DFS reports submitted by power reactor licensees in decommissioning.[5]

Results of the NRC Staff's Review—Operating Power Reactor Licensees

The NRC staff's review of the 2019 DFS reports for operating power reactor licensees resulted in the following findings:

- All 98 operating power reactor licensees met the reporting requirements of 10 CFR 50.75(f) and are currently demonstrating decommissioning funding assurance (DFA).

- As of the December 31, 2018 reporting period cutoff date, three operating power reactors with shortfalls were identified in the 2019 DFS review cycle (Beaver Valley Power Station, Unit 1 (BVPS, Unit 1); Clinton Power Station, Unit 1 (Clinton, Unit 1); and Perry Nuclear Power Plant, Unit 1 (PNPP)).

- According to its 2019 DFS report,[6] Exelon Generation Company, LLC (EGC), the licensee for Clinton, Unit 1, did not demonstrate DFA for this unit, as of December 31, 2018, due to market performance. However, according to EGC and verified by the NRC staff, as of February 28, 2019, DFA is demonstrated for Clinton, Unit 1, due to recovery in market performance.

- According to its 2019 DFS report,[7] FirstEnergy Nuclear Operating Company (FENOC), the licensee for BVPS, Unit 1 and PNPP, did not demonstrate DFA for either of these units, as of December 31, 2018. However, according to FENOC and verified by the NRC staff, as of January 31, 2019, DFA is demonstrated for PNPP, due to recovery in market performance. For BVPS, Unit 1, in both its 2019 DFS report and in a supplemental letter dated August 29, 2019,[8] related to a license transfer application for the FENOC reactor fleet, FENOC reported a shortfall in DFA. As a condition of its approval of the license transfer application on December 2, 2019,[9] the NRC required the applicants to implement and maintain a provisional trust agreement in the amount required to cover the BVPS, Unit 1 shortfall. Accordingly, DFA is demonstrated for BVPS, Unit 1.

- The 2017 DFS report review cycle included 100 operating power reactors. Since the last summary of staff review and findings for DFS reports,[10] two units have transitioned to a decommissioning status and are now included in the review of power reactor licensees in decommissioning.

- Amounts accumulated in the decommissioning trust funds for operating power reactors totaled approximately $56.5 billion as of December 31, 2018.

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4 ADAMS Accession No. ML19346E376
5 ADAMS Accession No. ML19346E377
6 ADAMS Accession No. ML19091A140
7 ADAMS Accession No. ML19074A242
8 ADAMS Accession No. ML19241A461
9 ADAMS Accession No. ML19303C953
10 ADAMS Accession No. ML18096B523

Results of the NRC Staff's Review—Power Reactor Licensees in Decommissioning

The NRC staff's review of the 2019 DFS reports for power reactor licensees in decommissioning resulted in the following findings:

- All 21 power reactor licensees in decommissioning met the reporting requirements of 10 CFR 50.82(a)(8)(v)–(vi).
- All 21 power reactor licensees in decommissioning demonstrated decommissioning funding assurance by either demonstrating a sufficient funding balance or by providing additional financial assurance to cover identified shortfalls.
- One of the 21 power reactor licensees in decommissioning reported a shortfall. In its submittal,[11] EGC, the licensee for Peach Bottom Atomic Power Station, Unit 1 (PBAPS, Unit 1), identified, and the NRC staff confirmed, a shortfall in funding for PBAPS, Unit 1, of about $15 million (in 2018 dollars). EGC provided additional financial assurance to cover the estimated cost to complete decommissioning at PBAPS, Unit 1, pursuant to 10 CFR 50.82(a)(8)(vi) and guidance in RG 1.159. Specifically, EGC indicated that collections from "non-bypassable charges"[12] from which EGC funds its decommissioning trust will be adjusted to cover any funding shortfall that exists. The NRC staff verified that the amounts to be collected will be adjusted, as necessary, in accordance with the applicable tariff in EGC's next filing to the Pennsylvania Public Utility Commission (PaPUC) of the Nuclear Decommissioning Cost Adjustment to cover any funding shortfall for PBAPS, Unit 1, at that time. The cost adjustment is made every five years pursuant to PaPUC Electric Tariff No. 4. The next effective date of a rate adjustment would be January 1, 2023. That scheduled adjustment provides additional assurance that funding will be available to complete radiological decommissioning at PBAPS, Unit 1.
- Current balances in the decommissioning trust funds for power reactor licensees in decommissioning totaled approximately $8.2 billion as of December 31, 2018.

CONCLUSION:

Based on its review of the 2019 DFS reports, the NRC staff finds that all licensees are in compliance with the decommissioning funding assurance reporting requirements of 10 CFR 50.75(f)(1)–(2) for operating power reactor licensees and 10 CFR 50.82(a)(8)(v)–(vi) for power reactor licensees in decommissioning. The staff also finds that all licensees are in compliance with the decommissioning funding assurance requirements of 10 CFR 50.75 and 10 CFR 50.82, as applicable, for the 2019 DFS reporting cycle.

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11 ADAMS Accession No. ML19091A140

12 The regulation at 10 CFR 50.2 states, "Non-bypassable charges mean those charges imposed over an established time period by a Government authority that affected persons or entities are required to pay to cover costs associated with the decommissioning of a nuclear power plant. Such charges include, but are not limited to, wire charges, stranded cost charges, transition charges, exit fees, other similar charges, or the securitized proceeds of a revenue stream."

COORDINATION:

The Office of the General Counsel has reviewed this paper and has no legal objection.

for J. Lubinski

John W. Lubinski, Director
Office of Nuclear Material Safety
and Safeguards

Enclosures:
1. 2019 DFS Report for Operating Power Reactor Licensees
2. 2019 DFS Report for Power Reactor Licensees in Decommissioning

SUBJECT: SUMMARY OF STAFF REVIEW AND FINDINGS OF THE 2019 DECOMMISSIONING FUNDING STATUS REPORTS FROM OPERATING AND DECOMMISSIONING POWER REACTOR LICENSEES

PKG - ML19346E375
SECY Paper - ML19346E378
Enclosure 1 - ML19346E376
Enclosure 2 - ML19346E377

***Concurred via email** **200800001**

OFFICE	NMSS/REFS	NMSS/REFS	NMSS/REFS
NAME	SHarwell	FMiller	KCoyne
DATE	12/12/2019	12/12/2019	12/16/2019
OFFICE	TechEd*	OGC – NLO*	NMSS
NAME	JDougherty	JWachutka	RLewis for JLubinski
DATE	12/16/2019	12/16/2019	12/31/2019

OFFICIAL RECORD COPY

2019 DECOMMISSIONING FUNDING STATUS REPORT
for Operating Power Reactor Licensees (December 31, 2018)

TABLE 1

Plant Name	Expected Shutdown Date as of 3/31/2019	Approx. No. of Years Remaining Before Expected Shutdown	Decommissioning Trust Fund (DTF) Balance (As of 12/31/18)	Projected DTF Balance[1] Before Decommissioning (2018$)	NRC Minimum[2] or Site-Specific Cost Estimate (SSCE[3]) (2018$)
Arkansas Nuclear One, Unit 1	05/20/2034	16	$506,719,075	$689,546,000	$472,331,427
Arkansas Nuclear One, Unit 2	07/17/2038	20	$405,329,792	$651,497,475	$491,386,711
Arnold (Duane) Energy Center	10/20/2020	2	$471,829,046	$462,395,253	$741,739,000 (SSCE)
Beaver Valley Power Station, Unit 1	05/31/2021	3	$286,891,783	$301,086,676	$748.559,222 (SSCE)
Beaver Valley Power Station, Unit 2	10/31/2021	3	$383,221,237	$405,545,049	$756,289,281 (SSCE)
Braidwood Station, Unit 1	07/29/2046	28	$344,387,000	$600,798,526	$516,910,976
Braidwood Station, Unit 2	10/17/2047	29	$373,111,000	$664,960,663	$516,910,976
Browns Ferry Nuclear Plant, Unit 1	12/20/2033	15	$382,129,027	$804,356,143	$670,652,094
Browns Ferry Nuclear Plant, Unit 2	06/28/2034	16	$372,441,358	$807,117,683	$670,652,094
Browns Ferry Nuclear Plant, Unit 3	07/02/2036	18	$337,644,437	$811,864,555	$670,652,094
Brunswick Steam Electric Plant, Unit 1	09/08/2036	18	$556,172,662	$792,968,330	$647,338,240
Brunswick Steam Electric Plant, Unit 2	12/27/2034	16	$612,128,747	$841,352,723	$647,338,240
Byron Nuclear Generating Station, Unit 1	09/16/2044	26	$378,722,000	$634,628,010	$516,910,976
Byron Nuclear Generating Station, Unit 2	08/02/2046	28	$364,942,000	$637,533,617	$516,910,976
Callaway Plant, Unit 1	10/18/2044	26	$516,590,664	$2,081,907,143	$943,465,000 (SSCE)
Calvert Cliffs Nuclear Power Plant, Unit 1	07/31/2034	16	$385,697,000	$526,609,517	$479,528,791
Calvert Cliffs Nuclear Power Plant, Unit 2	08/13/2036	18	$498,432,000	$709,461,610	$479,528,791
Catawba Nuclear Station, Unit 1	12/05/2043	25	$434,010,828	$808,514,773	$479,369,171
Catawba Nuclear Station, Unit 2	12/05/2043	25	$443,253,463	$833,206,989	$479,369,171
Clinton Power Station, Unit 1	09/29/2026	8	$543,165,000	$662,922,006	$681,913,929
Columbia Generating Station	12/20/2043	25	$267,400,000	$633,085,084	$560,620,749
Comanche Peak Nuclear Power Plant, Unit 1	02/08/2030	12	$509,817,614	$784,788,250	$407,782,271
Comanche Peak Nuclear Power Plant, Unit 2	02/02/2033	15	$570,766,848	$890,895,328	$407,782,271
Cooper Nuclear Station	01/18/2034	16	$600,371,186	$875,013,391	$635,296,272
Davis-Besse Nuclear Power Station, Unit 1	04/22/2037	19	$562,958,730	$812,054,756	$491,347,203
Diablo Canyon Power Plant, Unit 1	11/02/2024	6	$1,306,300,000	$2,642,507,129	$521,994,236
Diablo Canyon Power Plant, Unit 2	08/26/2025	7	$1,708,500,000	$2,759,198,842	$521,994,236
Donald C. Cook Nuclear Power Plant, Unit 1	10/25/2034	16	$648,808,262	$925,350,280	$512,358,221
Donald C. Cook Nuclear Power Plant, Unit 2	12/23/2037	19	$590,864,127	$904,956,927	$516,910,976
Dresden Nuclear Power Station, Unit 2	12/22/2029	11	$696,581,000	$866,390,450	$659,754,252
Dresden Nuclear Power Station, Unit 3	01/12/2031	13	$712,342,000	$906,892,241	$659,754,252
Farley (Joseph M.) Nuclear Plant, Unit 1	06/25/2037	19	$429,795,326	$693,130,020	$481,147,134
Farley (Joseph M.) Nuclear Plant, Unit 2	03/31/2041	23	$415,793,077	$724,773,434	$481,147,134

1 Includes growth from earnings and contributions.
2 Derived from minimum formula at Title 10 of the *Code of Federal Regulations* (10 CFR) 50.75(c). Incorporates labor, energy, and low-level waste (LLW) burial escalation factors.
3 Six licensees provided SSCEs.

ML19346E376

Enclosure 1

2019 DECOMMISSIONING FUNDING STATUS REPORT
for Operating Power Reactor Licensees (December 31, 2018)

TABLE 1

Plant Name	Expected Shutdown Date as of 3/31/2019	Approx. No. of Years Remaining Before Expected Shutdown	Decommissioning Trust Fund (DTF) Balance (As of 12/31/18)	Projected DTF Balance[1] Before Decommissioning (2018$)	NRC Minimum[2] or Site-Specific Cost Estimate (SSCE[3]) (2018$)
Fermi, Unit 2	03/20/2045	27	$1,290,000,000	$2,179,739,540	$1,124,206,329
Fitzpatrick (James A.) Nuclear Power Plant	10/17/2034	16	$837,714,000	$1,149,497,103	$656,818,742
Ginna (Robert E.) Nuclear Power Plant	09/18/2029	11	$453,696,000	$562,419,719	$447,772,783
Grand Gulf Nuclear Station, Unit 1	11/01/2044	26	$945,000,000	$1,619,813,428	$659,706,159
Hatch (Edwin I.) Nuclear Plant, Unit 1	08/06/2034	16	$556,872,142	$761,589,867	$642,017,733
Hatch (Edwin I.) Nuclear Plant, Unit 2	06/13/2038	20	$504,817,125	$752,073,127	$642,017,733
Hope Creek Generating Station, Unit 1	04/11/2046	28	$548,048,000	$946,314,050	$682,827,069
Indian Point Nuclear Generating, Unit 2	04/30/2020	2	$598,412,232	$665,712,399	$521,744,003
Indian Point Nuclear Generating, Unit 3	04/30/2021	3	$780,593,070	$885,909,858	$521,744,003
LaSalle County Station, Unit 1	04/17/2042	24	$510,017,000	$812,992,812	$681,913,929
LaSalle County Station, Unit 2	12/16/2043	25	$511,373,000	$841,358,445	$681,913,929
Limerick Generating Station, Unit 1	10/26/2044	26	$447,650,000	$970,726,285	$699,162,069
Limerick Generating Station, Unit 2	06/22/2049	31	$476,814,000	$1,189,138,805	$699,162,069
McGuire Nuclear Station, Unit 1	03/03/2041	23	$540,429,542	$843,022,670	$508,151,771
McGuire Nuclear Station, Unit 2	03/03/2043	25	$591,619,169	$960,505,217	$508,151,771
Millstone Power Station, Unit 2	07/31/2035	17	$672,500,000	$936,727,760	$471,737,576
Millstone Power Station, Unit 3	11/25/2045	27	$704,800,000	$1,206,886,636	$501,543,596
Monticello Nuclear Generating Plant, Unit 1	09/08/2030	12	$496,452,338	$867,609,095	$616,429,987
Nine Mile Point Nuclear Station, Unit 1	08/22/2029	11	$622,189,000	$770,007,040	$624,843,730
Nine Mile Point Nuclear Station, Unit 2	10/31/2046	28	$515,615,000	$899,252,314	$699,162,069
North Anna Power Station, Unit 1	04/01/2038	20	$454,380,000	$668,661,969	$488,174,147
North Anna Power Station, Unit 2	08/21/2040	22	$409,760,000	$631,781,904	$488,174,147
Oconee Nuclear Station, Unit 1	02/06/2033	15	$448,983,678	$595,906,645	$445,577,753
Oconee Nuclear Station, Unit 2	10/06/2033	15	$446,338,646	$600,340,911	$445,577,753
Oconee Nuclear Station, Unit 3	07/19/2034	16	$583,969,218	$797,319,523	$445,577,753
Palisades Nuclear Plant	05/31/2022	4	$443,630,000	$474,977,452	$480,360,545
Palo Verde Nuclear Generating Station, Unit 1	06/01/2045	27	$1,051,297,000	$1,785,294,634	$521,994,236
Palo Verde Nuclear Generating Station, Unit 2	04/24/2046	28	$1,099,314,000	$1,898,184,618	$521,994,236
Palo Verde Nuclear Generating Station, Unit 3	11/25/2047	29	$1,104,914,000	$1,969,184,363	$521,994,236
Peach Bottom Atomic Power Station, Unit 2	08/08/2033	15	$588,443,000	$846,583,161	$699,162,069
Peach Bottom Atomic Power Station, Unit 3	07/02/2034	16	$612,126,000	$903,857,122	$699,162,069
Perry Nuclear Power Plant, Unit 1	03/18/2026	8	$517,115,938	$597,734,336	$1,124,013,107 (SSCE)
Pilgrim Nuclear Power Station	05/31/2019	0	$1,027,714,005	$1,038,034,062	$1,187,994,231 (SSCE)

1 Includes growth from earnings and contributions.
2 Derived from minimum formula at Title 10 of the *Code of Federal Regulations* (10 CFR) 50.75(c). Incorporates labor, energy, and low-level waste (LLW) burial escalation factors.
3 Six licensees provided SSCEs.

TABLE 1

2019 DECOMMISSIONING FUNDING STATUS REPORT
for Operating Power Reactor Licensees (December 31, 2018)

Plant Name	Expected Shutdown Date as of 3/31/2019	Approx. No. of Years Remaining Before Expected Shutdown	Decommissioning Trust Fund (DTF) Balance (As of 12/31/18)	Projected DTF Balance[1] Before Decommissioning (2018$)	NRC Minimum[2] or Site-Specific Cost Estimate (SSCE[3]) (2018$)
Point Beach Nuclear Plant, Unit 1	10/05/2030	12	$401,729,516	$508,898,548	$447,201,839
Point Beach Nuclear Plant, Unit 2	03/08/2033	15	$378,522,034	$503,224,830	$447,201,839
Prairie Island Nuclear Generating Plant, Unit 1	08/09/2033	15	$492,616,045	$668,571,373	$441,873,225
Prairie Island Nuclear Generating Plant, Unit 2	10/29/2034	16	$461,002,122	$660,974,441	$441,873,225
Quad Cities Station, Unit 1	12/14/2032	14	$692,681,544	$926,844,344	$659,754,252
Quad Cities Station, Unit 2	12/14/2032	14	$747,179,957	$998,816,061	$659,754,252
River Bend Station, Unit 1	08/29/2045	27	$803,300,000	$1,589,990,378	$654,849,543
Robinson (H.B.) Steam Electric Plant, Unit 2	07/31/2030	12	$625,691,157	$788,656,382	$436,377,517
Salem Nuclear Generating Station, Unit 1	08/13/2036	18	$630,405,000	$966,669,058	$501,543,596
Salem Nuclear Generating Station, Unit 2	04/18/2040	22	$542,719,000	$907,501,292	$501,543,596
Seabrook Station, Unit 1	03/15/2050	32	$688,077,235	$1,282,885,897	$530,326,196
Sequoyah Nuclear Plant, Unit 1	09/17/2040	22	$211,311,189	$625,513,258	$508,151,771
Sequoyah Nuclear Plant, Unit 2	09/15/2041	23	$201,304,966	$626,380,349	$508,151,771
Shearon Harris Nuclear Power Plant, Unit 1	10/24/2046	28	$545,067,139	$950,617,972	$488,514,851
South Texas Project, Unit 1	08/20/2047	29	$459,285,587	$961,961,454	$407,782,271
South Texas Project, Unit 2	12/15/2048	30	$559,456,215	$1,171,141,534	$407,782,271
St. Lucie Plant, Unit 1	03/01/2036	18	$1,016,752,531	$1,435,232,094	$491,581,184
St. Lucie Plant, Unit 2	04/06/2043	25	$985,042,926	$1,601,901,726	$491,581,184
Summer (Virgil C.) Nuclear Station, Unit 1	08/06/2042	24	$299,517,198	$548,876,499	$458,916,086
Surry Power Station, Unit 1	05/25/2032	14	$456,600,000	$596,007,008	$473,140,598
Surry Power Station, Unit 2	01/29/2033	15	$457,800,000	$606,596,977	$473,140,598
Susquehanna Steam Electric Station, Unit 1	07/17/2042	24	$600,939,723	$962,725,845	$699,162,069
Susquehanna Steam Electric Station, Unit 2	03/23/2044	26	$661,493,829	$1,095,625,036	$699,162,069
Three Mile Island Nuclear Station, Unit 1	04/19/2019	0	$669,617,000	$909,702,208	$492,942,745
Turkey Point Nuclear Generating, Unit 3	07/19/2032	14	$839,232,304	$1,100,949,292	$475,568,111
Turkey Point Nuclear Generating, Unit 4	04/10/2033	15	$948,100,859	$1,262,550,210	$475,568,111
Vogtle Electric Generating Plant, Unit 1	01/16/2047	29	$351,543,613	$647,862,895	$508,151,771
Vogtle Electric Generating Plant, Unit 2	02/09/2049	31	$350,188,491	$672,787,932	$508,151,771
Waterford Steam Electric Station, Unit 3	12/18/2044	26	$481,644,236	$956,909,328	$508,151,771
Watts Bar Nuclear Plant, Unit 1	11/09/2035	17	$267,806,997	$622,872,985	$508,151,771
Watts Bar Nuclear Plant, Unit 2	10/21/2055	37	$101,186,523	$635,750,418	$508,151,771
Wolf Creek Generating Station, Unit 1	03/11/2045	27	$497,066,000	$1,173,254,740	$516,910,976

1 Includes growth from earnings and contributions.
2 Derived from minimum formula at Title 10 of the *Code of Federal Regulations* (10 CFR) 50.75(c). Incorporates labor, energy, and low-level waste (LLW) burial escalation factors.
3 Six licensees provided SSCEs.

2019 DECOMMISSIONING FUNDING STATUS REPORT

TABLE 2

for Power Reactor Licensees in Decommissioning (December 31, 2018)

Plant Name	Estimated Year of Completion of Radiological Decommissioning	Estimated Number of Years Remaining Until Part 50 License Termination	Decommissioning Trust Fund (DTF) Balance (As of 12/31/18) [1]	Estimated Remaining Cost to Complete Radiological Decommissioning (2018$)
Crystal River Nuclear Generating Plant, Unit 3	2073	55	$666,240,035	$746,689,950
Dresden Nuclear Power Station, Unit 1	2036	18	$342,623,000	$442,845,000
Fermi, Unit 1	2032	14	$22,800,000	$22,500,000
Fort Calhoun Nuclear Power Plant	2030	12	$975,633,000	$881,641,181
Humboldt Bay Power Plant, Unit 3	2019	1	$211,900,000	$24,200,000
Indian Point Nuclear Generating, Unit 1	2073	55	$471,200,000	$583,420,000
Kewaunee Power Station	2073	55	$574,411,000	$550,383,000
La Crosse Boiling-Water Reactor	2019	1	$21,700,000	$1,600,000
Millstone Power Station, Unit 1	2058	40	$504,610,000	$301,206,000
Nuclear Ship Savannah	2031	13	$108,000,000	$124,900,000
Oyster Creek Nuclear Generating Station	2035	17	$848,000,000	$618,000,000
Peach Bottom Atomic Power Station, Unit 1	2034	16	$117,728,000	$263,409,000
San Onofre Nuclear Generating Station, Unit 1	2030	12	$438,700,000	$77,300,000
San Onofre Nuclear Generating Station, Unit 2	2032	14	$1,497,800,000	$699,300,000
San Onofre Nuclear Generating Station, Unit 3	2032	14	$1,736,200,000	$688,800,000
Three Mile Island Nuclear Station, Unit 2	2053	35	$843,000,000	$1,320,506,000
Vallecitos Boiling-Water Reactor	2025	7	$11,992,513	$11,992,513
Vallecitos Experimental Superheat Reactor	2025	7	$15,646,541	$15,646,541
Vermont Yankee Nuclear Power Station	2073	55	$517,890,000	$498,450,000
Zion Nuclear Power Station, Unit 1	2020	2	Both Units Combined: $53,200,000	Both Units Combined: $24,000,000
Zion Nuclear Power Station, Unit 2	2020	2		

1 Dollar amounts reflected in the DTF Balance column may also include funding from other financial assurance methods, such as surety bonds and parent company guarantees, pursant to 10 CFR 50.75 (e)(1)(iii).

ML19346E377

Enclosure 2

Division of Spent Fuel Management
Interim Staff Guidance – 2, Revision 2

Issue: **Fuel Retrievability in Spent Fuel Storage Applications**

Introduction:

This Interim Staff Guidance (ISG) provides guidance to the staff for determining whether an application submitted under Title 10 of the *Code of Federal Regulations* (10 CFR) Part 72 (Ref. 1), “Licensing Requirements for the Independent Storage of Spent Nuclear Fuel, High-Level Radioactive Waste, and Reactor-Related Greater than Class C Waste” sufficiently demonstrates that the system is designed to allow ready retrieval of spent fuel. U.S. Nuclear Regulatory Commission (NRC) inspectors use the ISG and Inspection Procedures IP-60854 and IP-60855 (Ref. 2 and Ref. 3) during inspections to verify that licensees comply with 10 CFR 72.122(l). This ISG does not apply to submitted applications seeking approval under 10 CFR Part 71, “Packaging and Transportation of Radioactive Material” (Ref. 4). This guidance is not a regulation or a requirement as it addresses options to meet the regulation. Additionally, applicants may propose alternate methods to comply with the regulation which would be evaluated on a case-by-case basis. A background section is included in Appendix A.

Regulatory Basis

The regulations for safe storage of spent nuclear fuel for licensees are in 10 CFR Part 72. Retrievability is specifically mentioned in 10 CFR 72.122(l), which states that “storage systems must be designed to allow ready retrieval of spent fuel, high-level radioactive waste, and reactor-related greater than class C waste for further processing or disposal.” The NRC interprets this regulation to require that a storage system be designed to allow for ready retrieval in the initial design, amendments to the design, and in license renewal, through the aging management of the design. Retrievability is applicable only during normal and off-normal conditions; it does not apply to accident conditions (Ref. 5). The retrievability requirement applies to all general licensed and specific licensed independent spent fuel storage installations (ISFSIs), including wet storage ISFSIs, however most of current licensed ISFSIs use only dry storage. 10 CFR 72.236(m) states that certificate of compliance (CoC) holders should design for retrievability; “[t]o the extent practicable in the design of spent fuel storage casks, consideration should be given to compatibility with removal of the stored spent fuel from a reactor site, transportation, and ultimate disposition by the Department of Energy.”

Applicability:

The staff will apply ISG-2, Rev. 2 in reviewing ISFSI applications conducted in accordance with NUREG-1536, "Standard Review Plan for Dry Cask Storage Systems" (Ref. 6), NUREG-1567, "Standard Review Plan for Spent Fuel Dry Storage Facilities" (Ref. 7), or NUREG-1927, “Standard Review Plan for Renewal of Specific Licenses and Certificates of Compliance for Dry Storage of Spent Nuclear Fuel” (Ref. 8 and 9).

This revision of ISG-2 redefines retrievability and supersedes the definition of retrievability in NUREG-1536, NUREG-1567, and NUREG-1927 and applicable storage

ISGs. The previous revision of ISG-2, Rev. 1 (Ref. 10) is superseded in its entirety by ISG-2, Rev. 2.

Technical Review Guidance

ISG-2, Rev. 2 defines ready retrieval as "the ability to safely remove the spent fuel from storage for further processing or disposal." In order to demonstrate the ability for ready retrieval, a licensee should demonstrate it has the ability to perform any of the three options below. These options may be utilized individually or in any combination or sequence, as appropriate.

A. remove individual or canned spent fuel assemblies from wet or dry storage,
B. remove a canister loaded with spent fuel assemblies from a storage cask/overpack,
C. remove a cask loaded with spent fuel assemblies from the storage location.

The NRC's licensing reviews and inspection oversight of the design, fabrication, construction, and operation of an ISFSI, assures the requirements of 10 CFR Part 72, including retrievability, are maintained during the initial storage period. When spent fuel is stored beyond the initial NRC-approved period of operation, 10 CFR 72.42 requires a licensee renew its storage license. Applications for renewal must contain revised technical requirements and operating conditions (fuel storage, surveillance and maintenance, and other applicable 10 CFR Part 72 requirements) that address aging mechanisms and aging effects that could affect structures, systems, and components (SSCs) relied upon for the safe storage of spent fuel. The renewal application must include (1) time-limited aging analyses (TLAAs), if applicable, that demonstrate that SSCs important to safety will continue to perform their intended function for the requested period of extended operation, and (2) aging management programs (AMPs) for management of issues associated with aging that could adversely affect SSCs important to safety.

In verifying that all applicants for an initial ISFSI license or an ISFSI license amendment meet the retrievability requirement of 10 CFR 72.122(l), the reviewer must find there is reasonable assurance the storage system design allows for ready retrieval by the use of option A, B, or C or a combination of the options. A dry storage system may demonstrate retrievability by the use of known and controlled fuel selection, limits on the loading temperature, known atmospheric environment, and transfer cask or canister temperature control (Ref. 11 and 12). The reviewer should also verify that applications for all storage systems identify the SSCs important to safety and the SSC subcomponents that are relied upon for ready retrieval. The reviewer should further verify that the Technical Specifications (TSs) included in the application provide for the maintenance of SSCs relied upon for ready retrieval. The revised definition of retrievability does not obviate the need for appropriate control of parameters during loading, vacuum drying, and transfer to the storage location (e.g., dry storage pad).

When an applicant for an initial ISFSI license or an applicant for an amendment to an ISFSI license relies on Option A to demonstrate ready retrieval, the reviewer should confirm that the applicant demonstrated the fuel assemblies will not exhibit gross degradation, and will be removable. Additional review will be needed in the case where there is an assembly with gross degradation or an assembly contains rods with breaches greater than a pinhole leak or a hairline crack (i.e., gross ruptures that could lead to release of fuel particulates per ISG-1, Rev. 2 [Ref. 12]). The reviewer should confirm

that the applicant demonstrates the fuel assembly can be placed inside a secondary container, as described in ISG-1 as a "can for damaged fuel." The secondary container must confine the fuel particulate to a known volume and be capable of removal.

If an applicant for an initial dry storage ISFSI license or an applicant for an ISFSI license amendment relies upon Option A to demonstrate ready retrieval, it is likely the storage cask/canister will, at some point, need to be moved from the storage location to a location where the spent fuel assemblies can be removed from the cask/canister. When the reviewer anticipates that the cask/canister will have to be moved, the reviewer should confirm the applicant relying upon Option A to demonstrate ready retrieval, also demonstrates ready retrieval under Option B or Option C. This is consistent with the previous guidance on fuel retrievability.

When an applicant for an initial ISFSI license or for an ISFSI license amendment demonstrates ready retrieval with Option B or Option C, the continued ready retrieval of the storage system should be addressed in its TS. However, in addition to the TS, an applicant may also propose to implement a program to identify, monitor, and mitigate possible degradation that could impact the intended function of the dry storage system's SSCs and subcomponents of the dry storage system, that are relied upon to comply with the retrievability requirements.

The NRC reviewer of an application for renewal of an ISFSI license should verify the 10 CFR 72.122(l) retrievability requirement is met, by ensuring that the approved design bases for the item being relied upon in the option(s) chosen (e.g., fuel assembly, cask, or canister) to demonstrate ready retrieval, including any programs implemented, has not been altered. Additionally, the reviewer should verify that the AMPs and TLAAs provide reasonable assurance that the approved design bases will be maintained during the period of extended operation. This will include reviewing operating experience, including inspections and analyses performed during the initial storage period for ensuring SSCs relied upon for ready retrieval were maintained. The reviewer should refer to Draft NUREG-1927, Rev. 1 (Ref. 8) for additional guidance.

CoC holders and applicants for a CoC are not required by regulation to demonstrate retrievability under 10 CFR 72.122(l); however, 10 CFR 72.236(m), which applies to CoC holders, states that retrievability should be considered to the extent practicable in the design to consider removal of the spent fuel from storage, transportation, and ultimate disposition. When a CoC applicant for an initial certificate, amendment, or revision chooses to incorporate retrievability aspects, the reviewer should confirm the retrievability aspects are technically justified and verify that Part 72 requirements affected by retrievability are evaluated and met. This may include the NRC reviewer confirming that the design for the dry storage system includes an evaluation for potential degradation mechanisms for both the storage cask/canister and the spent fuel to assure that the design of the system has considered removal of the spent fuel from storage during the storage term. Note that the general licensee must comply with the retrievability requirement in 10 CFR 72.122(l), and should demonstrate that canister/casks meet the amendment loading requirements.

Recommendation:

This ISG recommends the following definition to be used by staff when evaluating Part 72 applications:

> Ready retrieval: The ability to safely remove the spent fuel from storage for further processing or disposal.
>
> Acceptable means for removing the spent fuel from storage include the ability to perform any of the three options below. These options may be utilized individually or in any combination or sequence, as appropriate. :
>
> A. remove individual or canned spent fuel assemblies from wet or dry storage,
> B. remove a canister loaded with spent fuel assemblies from a storage cask/overpack,
> C. remove a cask loaded with spent fuel assemblies from the storage location.

The staff recommends the definitions for ready retrieval be incorporated into NUREG-1536, NUREG-1567, and NUREG-1927. These definitions do not necessitate any actions for currently approved storage systems.

Approved:	**/RA/**	**4/26/16**
	Mark Lombard, Director	**Date**
	Division of Spent Fuel Management	

Appendix A

This Appendix is provided to give insight on the history and evolution of the regulatory requirement of fuel retrievability.

Section 141(b)(1)(C) of the Nuclear Waste Policy Act (NWPA) of 1982, as amended (Ref. 13), requires that each monitored retrievable storage (MRS) facility be designed "…to provide for the ready retrieval of such spent fuel and waste for further processing or disposal." The Nuclear Regulatory Commission (NRC) codified this portion of the NWPA in its 1988 final rulemaking "Licensing Requirements for the Independent Spent Fuel Storage of Spent Nuclear Fuel and High-Level Radioactive Waste" (Ref. 14), which added MRSs to the scope of Title 10 of the *Code of Federal Regulations* (10 CFR) Part 72 and required retrievability for all independent spent fuel storage installations (ISFSIs), 10 CFR 72.122(l).

For general and specific licensees, the regulation regarding retrievability is 10 CFR 72.122(l), which requires that "storage systems must be designed to allow ready retrieval of spent fuel, high-level radioactive waste, and reactor-related greater than class C waste for further processing or disposal." It is supported by 10 CFR 72.122 (h)(1), which requires that, for confinement barriers and systems, "The spent fuel cladding must be protected during storage against degradation that leads to gross ruptures or the fuel must be otherwise confined such that degradation of the fuel during storage will not pose operational safety problems with respect to its removal from storage. This may be accomplished by canning of consolidated fuel rods or unconsolidated assemblies or other means as appropriate." 10 CFR 72.236(m) directs that certificate of compliance (CoC) holders and applicants consider retrievably in cask design. The regulation states that, "[t]o the extent practicable in the design of spent fuel storage casks, consideration should be given to compatibility with removal of the stored spent fuel from a reactor site, transportation, and ultimate disposition by the Department of Energy."

Additionally, the NRC has previously recognized that "in the interest of decreasing radiation exposures, storage casks should be designed to be compatible with transportation and Department of Energy [DOE] design criteria to the extent practicable… to the extent that cask designers can avoid return of the spent fuel from dry cask storage to reactor basins for transfer to a transport cask before moving it off site for disposal" (Ref. 15).

The NRC staff's previous position on retrievability, as stated in interim staff guidance (ISG) - 2, Rev. 1 (Ref. 10), defined ready retrieval as "the ability to move a canister containing spent fuel to either a transportation package or to a location where the spent fuel can be removed. Ready retrieval also means maintaining the ability to handle individual or canned spent fuel assemblies by the use of normal means."

The guidance for retrievability in ISG-2, Rev. 1 was developed when an operating repository was expected to be operating in the near future. As of 2015, the duration of the storage of spent fuel storage at an ISFSI or MRS remains uncertain. Therefore, the staff re-assessed the regulatory necessity and practical impact of maintaining and confirming the ability to handle an individual fuel assembly from the canister or cask by normal means as part of the guidance on retrievability.

The NRC's licensing reviews and inspection oversight of the design, fabrication, construction, and operation of an ISFSI, assures that the safety and retrievability requirements of 10 CFR Part 72 are maintained during the initial storage period. When spent fuel storage will continue beyond the initial NRC-approved period of operation, the NRC's storage regulations that 10 CFR 72.240 require that renewal applications contain revised technical requirements and operating conditions (fuel storage, surveillance and maintenance, and other Part 72 requirements) that address aging mechanisms and aging effects that could affect structures, systems, and components (SSCs) relied upon for the safe storage of spent fuel. The renewal application must include (1) time-limited aging analyses (TLAAs), if applicable, that demonstrate that SSCs important to safety will continue to perform their intended function for the requested period of extended operation, and (2) aging management programs (AMPs) for management of issues associated with aging that could adversely affect SSCs important to safety.

Under the guidance of ISG-2, Rev. 1, if a licensee's ability to demonstrate ready retrieval relies on the handling of each individual fuel assembly from a canister or cask by normal means, then periodic monitoring or inspections may be required to verify the condition of the fuel and the internal components of the storage system. Because of the difficulties in accessing the spent fuel and the interior components of some storage systems, opening the storage system may be necessary to conduct inspection, monitoring, and remediation. Opening a storage system is labor intensive, but more importantly, it exposes workers to additional dose, and particularly for welded canisters, may require breaching and reestablishing the confinement boundary with no additional safety benefit. Additionally, it is not current practice to open the storage system to verify fuel condition.

Consistent with the staff's ongoing review of the regulatory framework for spent fuel storage and transportation (see COMSECY-10-0007, Ref. 16), the staff began exploring alternatives to the guidance on the application of ready retrieval. The staff's review has centered on redefining the ability of the fuel assemblies to be removed from a canister or cask by normal means, but maintaining the ability of the canister or cask to be removed from the storage location. By redefining guidance on the ability to remove the individual spent fuel assemblies or canned assemblies by normal means and providing alternatives, the spent fuel would still be retrieved safely and be readied for transportation consistent with the law and regulations. In addition this approach assures that the confinement of spent fuel in dry storage is maintained without the potential negative impacts that could may accompany opening the storage system.

In an effort to engage stakeholders in this discussion and solicit stakeholder views, the staff held two public meetings on July 27, 2011 and August 16, 2012 (Ref. 17 and 18). Additionally, in January 2013, NRC issued a *Federal Register* notice (Ref. 19) requesting public comment on several topics, including retrievability. The NRC received 18 sets of comments on the *Federal Register* notice (Ref. 20). Staff work in this area was delayed until recently due to work on the storage renewal regulatory framework and high burnup fuel related activities. For this reason, the staff held an additional public meeting on July 29, 2015, to provide an update on the staff's work on retrievability (Ref. 21).

In addition to conducting the public dialogue, the staff considered the methods used in other countries for the dry storage of spent nuclear fuel and reviewed international guidance for spent fuel storage. The staff participated in several multilateral working groups related to extended spent fuel storage. The staff reviewed the International Atomic Energy Agency's (IAEA) Specific Safety Guide No. SSG-15, "Storage of Spent

Nuclear Fuel" (Ref. 22). This IAEA guide is consistent with the NRC's current position that spent fuel should be retrievable under normal and off-normal design conditions. The revision of ISG-2, Rev. 2 does not change this view. The IAEA's guidance states retrievability is also applicable during accident conditions, which differs from the NRC's position (Ref. 5).

This updated guidance, ISG-2, Rev. 2, presents a practical approach for implementation of fuel retrievability that will continue to protect public health and safety while reducing the negative impacts associated with the approach established in ISG-2, Rev.1.

References:

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2. U.S. Nuclear Regulatory Commission. 2008. "Preoperational Testing of an Independent Spent Fuel Storage Installations," Inspection Procedure-60854, Washington DC. Agencywide Document Access and Management System (ADAMS) Accession No. ML073100468.

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4. 10 CFR Part 71. *Code of Federal Regulations*, Title 10, Energy, Part 71, "Packaging and Transportation of Radioactive Material." Washington, DC.

5. U.S. Nuclear Regulatory Commission. 1998. "Post Accident Recovery and Compliance with 10 CFR 72.122(l)," Spent Fuel Storage and Transportation ISG-3, Washington, DC. ADAMS Accession No. ML092800380.

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16. U.S. Nuclear Regulatory Commission. 2010. COMSECY-10-0007: "Project Plan for the Regulatory Program Review to Support Extended Storage and Transportation of Spent Nuclear Fuel." ADAMS Accession No. ML101390216.

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18. U.S. Nuclear Regulatory Commission. 2012. "Summary of the August 16-17, 2012 Meeting to Obtain Stakeholder Feedback on Enhancements to the Licensing and Inspection Programs for Spent Fuel Storage and Transportation Under 10 CFR Parts 71 and 72," ADAMS Accession No. ML12261A069.

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20. U.S. Nuclear Regulatory Commission. 2015. Package of Public Comments: "Public Comments on *Federal Register* Notice on Retrievability and Cladding Integrity," ADAMS Accession No. ML15110A370.

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION REPORT

Docket No. 72-1040
HI-STORM UMAX Canister Storage System
Holtec International, Inc.
Certificate of Compliance No. 1040

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PRELIMINARY SAFETY EVALUATION REPORT
Docket No. 72-1040
HI-STORM UMAX Canister Storage System
Holtec International, Inc.
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1 SUMMARY

By letter dated June 29, 2012, as supplemented July 16, November 20, 2012, and January 30, April 2, April 19, June 21, August 28, December 6, December 31, 2013, and January 13, and 28, 2014, Holtec International (Holtec) submitted an application to the U.S. Nuclear Regulatory Commission (NRC) for the HI-STORM UMAX Canister Storage System, Certificate of Compliance (CoC) No. 1040. The proposed application intends to provide an underground storage option compatible with the Holtec HI-STORM Flood/Wind (FW) Multipurpose Canister (MPC) System, CoC No. 1032.

This safety evaluation report (SER) documents the review and evaluation of the proposed application. The SER uses the same section-level format provided in NUREG-1536, Revision 1, “Standard Review Plan for Dry Cask Storage Systems,” with some differences implemented for clarity and consistency. The NRC staff (staff) followed the guidance of NUREG-1536, Revision 1, Interim Staff Guidance (ISG) -11, “Cladding Considerations for the Transportation and Storage of Spent Fuel” and ISG-21, “Use of Computational Modeling Software” in performing its regulatory evaluation. Unless specifically identified, staff findings and conclusions have been made using these guidance documents as the bases of determination.

The staff’s assessment is to determine that CoC No. 1040, meets the applicable requirements of 10 CFR Part 72 for independent storage of spent fuel and of 10 CFR Part 20 for radiation protection.

1.1 General Information Evaluation

The objective of the review of the general information evaluation of the HI-STORM UMAX Canister Storage System is to ensure that Holtec has provided a description that is adequate to familiarize reviewers and other interested parties with the pertinent features of the system.

1.2 HI-STORM UMAX CANISTER STORAGE SYSTEM GENERAL DESCRIPTION AND OPERATIONAL FEATURES

In Section 1.2 of the FSAR, the applicant provides the general description and operational features of the this system. According to the applicant, the HI-STORM (acronym for Holtec International Storage Module) UMAX Canister Storage System is a spent nuclear fuel storage

system designed to be in full compliance with the requirements of 10 CFR Part 72. The model designation "UMAX" denotes underground – maximum capacity. The proposed application intends to provide an underground storage option compatible with the Holtec HI-STORM Flood/Wind (FW) System as described in the HI-STORM FW Final Safety Analysis Report (FSAR). The underground structure system is described in the HI-STORM UMAX Canister Storage System FSAR. Unless designated otherwise in this SER the term “FSAR” denotes the HI-STORM UMAX Canister Storage System FSAR.

The HI-STORM UMAX Canister Storage System stores a hermetically sealed canister containing spent nuclear fuel (SNF) in an in-ground vertical ventilated module (VVM). The HI-STORM UMAX Canister Storage System is designed to provide long-term underground storage of loaded multi-purpose canisters (MPC) previously certified for storage in CoC No. 1032. The HI-STORM UMAX VVM is the underground equivalent of the HI-STORM FW storage module. Although the storage cavity dimensions and the air ventilation system in the HI-STORM UMAX VVM have been selected to enable it to also store all MPCs certified for storage in the HI-STORM 100 storage module, the proposed CoC No. 1040 does not seek to support the certification of all MPCs certified for storage in the HI-STORM 100 storage module at this time. The applicant explains that safety analyses and evaluations of the HI-STORM 100 MPCs under storage in HI-STORM UMAX are nevertheless included in HI-STORM UMAX FSAR, as appropriate, to provide a comparative reference for the licensing-basis analyses of the HI-STORM FW canisters (MPC-37 & MPC-89).

The applicant states that the HI-STORM UMAX Canister Storage System can store either PWR or BWR fuel assemblies, in the MPC-37 or MPC-89, respectively. The number associated with the MPC is the maximum number of fuel assemblies the MPC can contain in the fuel basket. The external diameters of the MPC-37 and MPC-89 are identical to allow the use of a single storage module design, however the height of the MPC, as well as the storage module and transfer cask, are variable based on the SNF to be loaded.

According to the applicant, the HI-STORM UMAX Canister Storage System is autonomous in-as-much as it provides SNF and radioactive material confinement, radiation shielding, criticality control and passive heat removal independent of any other facility, structures, or components at the site. The surveillance and maintenance operations of the HI-STORM UMAX Canister Storage System are minimized since the system is completely passive and is composed of proven materials. The HI-STORM UMAX Canister Storage System can be used either singly or as an array at an independent spent fuel storage installation (ISFSI). The site for an ISFSI can be located either at a nuclear reactor facility or an away-from-reactor location.

1.3 Staff Evaluation Findings

F1.1 The general description and discussion of the HI-STORM UMAX Canister Storage System is acceptably presented in Section 1.2 of the FSAR. Special attention to design and operating characteristics, unusual or novel design features, and principal considerations important to safety (ITS) have been acceptably provided.

F1.2 Drawings for SSCs ITS are presented in Section 1.5 of the FSARs in sufficient detail for the staff to provide sound regulatory findings. A listing of those drawings (including dates and revision numbers) that were relied upon as a basis for approval appears in Section 1.5 of the FSARs.

F1.3 Specifications for the SNF to be stored in the HI-STORM UMAX Canister Storage System are acceptably provided in the FSAR Section 2.1.

F1.4 The quality assurance program and implementing procedures are acceptably described in Section 1.3 of the FSAR.

F1.5 The HI-STORM UMAX Canister Storage System is not being certified under 10 CFR Part 71 for use in transportation.

The staff concludes that the information presented in Chapter 1, "General Information" of the SAR satisfies the requirements for the general description under 10 CFR Part 72. This finding is reached on the basis of a review that considered NUREG 1536, Rev. 1.

2 PRINCIPAL DESIGN CRITERIA EVALUATION

The objective of evaluating the principal design criteria related to SSCs that are ITS is to ensure that they comply with the relevant general criteria established in 10 CFR Part 72. The staff specifically reviewed principal design criteria to determine with reasonable assurance that all design criteria are addressed in the FSAR. The following areas of review were specifically reviewed by the staff:

- Structures, Systems, and Components Important to Safety
- Design Basis for Structures, Systems, and Components Important to Safety
- Spent Nuclear Fuel (SNF) Specifications
- External Conditions
- Design Criteria for Safety Protection Systems
- Structural
- Thermal
- Shielding/Confinement/Radiation Protection
- Criticality
- Material Selection
- Operating Procedures
- Acceptance Tests and Maintenance
- Decommissioning

2.1 Structures, Systems and Components Important to Safety

HI-STORM UMAX Canister Storage System SSCs that are ITS are acceptably identified in Chapter 2 of the HI-STORM UMAX Canister Storage System and the HI-STORM FW System FSARs. SER section 3.2.1 provides a description of the major components described in the FSARS and the correlation between the FSARS. The safety classifications are based on the guidance in U.S. Nuclear Regulatory Commission, "Classification of Transportation Packaging and Dry Spent Fuel Storage System Components According to Importance to Safety," NUREG/CR-6407, INEL-95/0551, February 1996, and per NUREG 1536, Rev. 1, and are therefore acceptable.

2.2 Design Basis for Structures, Systems and Components Important to Safety

The HI-STORM UMAX Canister Storage System design criteria summary acceptably includes the allowed range of spent fuel configurations and characteristics, the enveloping conditions of use, and the bounding site characteristics.

2.2.1 Spent Fuel Specifications

According to the FSAR, the HI-STORM UMAX Canister Storage System is designed to store up to either 37 PWR fuel assemblies or up to 89 BWR fuel assemblies. Detailed specifications for the approved fuel assemblies are provided in the HI-STORM FW FSAR Section 2.1. These include the maximum enrichment, maximum decay heat, maximum fuel assembly average burnup, minimum cooling time, maximum initial enrichment, and detailed physical fuel assembly parameters. The limiting fuel specifications are based on the fuel parameters considered in the structural, thermal, shielding, criticality and confinement analyses.

2.2.2 External Conditions

The HI-STORM UMAX Canister Storage System FSAR Section 2.2 identifies the bounding site environmental conditions and natural phenomena for which the HI-STORM UMAX Canister Storage System is analyzed.

2.3 Design Criteria for Safety Protection Systems

The principal design criteria for the HI-STORM UMAX Canister Storage System are acceptably identified in the HI-STORM UMAX Canister Storage System and HI-STORM FW System FSARs, Chapter 2.

2.4 Staff Evaluation Findings

F2.1 The FSAR and docketed materials adequately identify and characterize the SNF to be stored in the DSS in conformance with the requirements given in 10 CFR 72.236.

F2.2 The FSAR and the docketed materials relating to the design bases and criteria meet the general requirements as given in 10 CFR 72.122(a), (b), (c), (f), (h)(1), (h)(4), (i), and (l).

F2.3 The FSAR and docketed materials relating to the design bases and criteria for structures categorized as important to safety meet the requirements given in 10 CFR 72.122(a), (b)(1), (b)(2) and (b)(3), (c), (f), (h)(1), (h)(4), and (i); and 10 CFR 72.236.

F2.4 The FSAR and docketed materials meet the regulatory requirements for design bases and criteria for thermal consideration as given in 10 CFR 72.122 (a), (b)(1), (b)(2) and (b)(3), (c), (f), (h)(1), (h)(4), and (i).

F2.5 The FSAR and docketed materials relating to the design bases and criteria for shielding, confinement, radiation protection, and ALARA considerations meet the regulatory requirements as given in 10 CFR 72.104(a) and (b); 10 CFR 72.106(b); 10 CFR 72.122(a), (b), (c), (f), (h)(1), (h)(4), and (i); 10 CFR 72.126(a).

F2.6 The FSAR and docketed materials relating to the design bases and criteria for criticality safety meet the regulatory requirements as given in 10 CFR 72.124(a) and (b).

F2.7 The FSAR and docketed materials relating to the design bases and criteria for retrievability meet the regulatory requirements as given in 10 CFR 72.122(a), (b)(1), (b)(2), and (b)(3), (c), (f), (h)(1), (h)(4), and (l).

F2.8 The FSAR and docketed materials relating to the design bases and criteria for other SSCs not important to safety but subject to NRC approval meet the general regulatory requirements as given in the following subparts of 10 CFR Part 72: Subpart E, “Siting Evaluation Factors” 72.104 and 72.106; Subpart F, “General Design Criteria” 72.122, 72.124, and 72.126; and Subpart L, “Approval of Spent Fuel Storage Casks.”

The staff finds that the principal design criteria for the HI-STORM UMAX Canister Storage System are acceptable with regard to demonstrating compliance with the regulatory requirements of 10 CFR Part 72. This finding is based on a review that considered the regulation itself, NUREG 1536, Rev. 1, applicable codes and standards, and accepted engineering practices. More detailed evaluations of design criteria and assessments of compliance with those criteria are presented in SER Sections 3, 4,5,6,7, and 8.

3 STRUCTURAL EVALUATION

3.1 Overview

In this portion of the dry storage system (DSS) review, the NRC evaluates aspects of the DSS design and analysis related to structural performance under normal and off-normal operations, accident conditions, and natural phenomena events. In conducting this evaluation, the NRC staff seeks a high degree of assurance that the cask system will maintain confinement, subcriticality, radiation shielding, and retrievability or recovery of the fuel, as applicable, under all credible loads for normal and off-normal conditions accidents, and natural phenomenon events.

The objective of the structural review is to assess the safety analysis of the structural design features, the structural design criteria, and the structural analysis and evaluation criteria used to confirm the structural performance of the HI-STORM UMAX Canister Storage System under normal operations, off-normal operations, accident conditions and natural phenomena events for those ITS SSCs.

The review was conducted utilizing applicable regulations in 10 CFR 72.124 (a), 72.234 (a) and (b), 72.236 (b), (c), (d), (g), (h), and (l) that identify the specific requirements for spent fuel storage cask approval and fabrication.

3.2 Structural Design

3.2.1 Overview

As described in FSAR section 1.2, the HI-STORM UMAX Canister Storage System has three major components: MPC-37 and MPC-89, the HI-TRAC VW transfer cask, and the HI-STORM UMAX VVM. The MPCs and the HI-TRAC components used in the HI-STORM UMAX Canister Storage System are identical to those reviewed and approved in the HI-STORM FW System, CoC No. 72-1032. No other approvals were sought for MPC variants for this licensing action.

The structural sub-components of the HI-STORM UMAX VVM include the following items: the steel and concrete closure lid, the steel cavity enclosure container (CEC) shell, the independent spent fuel storage installation (ISFSI) pad, the support foundation pad (SFP), the subgrade, the under-grade, and an optional enclosure wall.

All components classified as ITS are designated on the licensing drawings in FSAR section 1.5.

3.2.1.1 VVM Components and ISFSI Structure

The FSAR states that the HI-STORM UMAX VVM serves as a missile and radiation barrier, provides flow paths for natural convection, and provides kinematic stability to the system. The VVM is not a pressure vessel since it is open to the environment. Each subcomponent is summarized below based upon information in the FSAR:

CEC – A thick walled open top shell welded to a bottom base plate that defines the storage cavity for the MPCs. The CEC rests on the SFP and is surrounded laterally by a self-hardening engineered subgrade.

Closure Lid - A steel structure filled with plain concrete that is designed to protect the VVM from the impact of the design basis missiles as well as provide an inlet and outlet for air flow.

ISFSI Pad - A reinforced concrete slab that surrounds the upper portion of the CEC and extends to the underside of the CEC Flange. The ISFSI pad provides robust support for a loaded transporter and to enable rainwater to flow away from the storage array.

SFP - A reinforced concrete provides below grade support to the CEC for loadings due to seismic events and long term settlement.

Subgrade and Under-grade - The soil between the SFP and the ISFSI pad and lateral to the CECs which is replaced with a self-hardening engineered subgrade (SES) is the subgrade. The undisturbed soil in the space below the SFP is referred to as the under-grade.

Enclosure Wall (optional) - The Enclosure Wall was designed to provide a barrier to the engineered fill beneath the ISFSI pad such that each VVM array would be distinct from surrounding soil or other VVM arrays. Another function of the Enclosure Wall is to provide a means of preventing water intrusion beneath the ISFSI pad.

3.2.1.2 Multi-Purpose Canisters

As described in the FSAR, the HI-STORM UMAX system utilizes two MPCs as confinement vessels: the MPC-37 for pressurized water reactor (PWR) fuel and the MPC-89 for boiling water reactor (BWR) fuel. These MPCs have been previously reviewed and approved for storage (CoC No. 1032) and all relevant evaluations are presented in the HI-STORM FW FSAR. Only relevant information necessary to evaluate the interaction between the HI-STORM UMAX VVM and the MPCs was presented in the HI-STORM UMAX Canister Storage System application.

3.2.1.3

3.2.1.4 Transfer cask (HI-TRAC VW)

According to the FSAR, the HI-STORM UMAX Canister Storage System utilizes the HI-TRAC VW transfer cask to provide a missile and radiation barrier during transport of the MPCs from the fuel pool to the HI-STORM UMAX VVM. The HI-TRAC VW has been previously reviewed and approved for storage activities (CoC No. 1032) and all relevant evaluations are presented in the HI-STORM FW FSAR. Only relevant information necessary to evaluate the interaction between the HI-STORM UMAX VVM and the HI-TRAC was presented in the HI-STORM UMAX Canister Storage System application.

3.2.2 Design Criteria and Applicable Loads

Table 2.3.1 of the FSAR summarizes all loads, design criteria, applicable regulations, reference codes and standards for the VVM.

Table 2.3.2 of the FSAR summarizes design data for HI-STORM UMAX Canister Storage System.

3.2.2.1 Applicable Loadings

Loadings applicable to the HI-STORM UMAX Canister Storage System are defined in FSAR Sections 2.4 and 2.5.

3.2.2.2 Design Basis Loads and Load Combinations

Table 2.4.1 of the FSAR contains design basis loads and acceptance criteria applicable to VVM components.

Table 2.4.3 of the FSAR contains load combinations applicable to ISFSI structures.

3.2.2.3 Allowable Stresses

The ITS components of the HI-STORM UMAX system are identified on the design drawings in FSAR Section 1.5. Allowable stresses and stress intensities for American Society of Mechanical Engineers (ASME) B&PV Code (Code) are identified in Tables 3.1.11 and 3.1.12 of the FSAR. Tables 3.1.2 to 3.1.8 of the FSAR contain tabulated values for all VVM and MPC components. Specifically, FSAR Table 3.1.4 contains Level A allowable stresses, FSAR Table 3.1.5 contains Level B allowable stresses, and FSAR Table 3.1.6 contains Level D allowable stresses.

3.2.3 Stress Analysis Models and Computer Codes

The applicant's finite element analysis was performed with LS-DYNA, which is a commercial explicit dynamics code for dynamic events, and ANSYS which was used for static stress analysis. Some stress analysis was also performed with closed form classical methods.

3.2.3.1 HI-STORM UMAX VVM

The applicant's VVM finite element model was developed for the seismic soil structure interaction (SSI) analysis of a representative 5×5 VVM array loaded with the tallest and heaviest approved MPC.

Key features of the applicant's FEA model included:

- Shell elements were used for divider shell and CEC

- Thick shell and solid elements were used for the CEC base plate and MPC pedestals, respectively.

- VVM lid was simplified to a rigid solid body to maximize the contact forces.

- The bounding MPC was modeled in the VVM as a rigid cylinder which yielded bounding stresses in the VVM and bounding loads for the ISFSI structures.

- FEA meshing developed during the licensing process for the HI-STORM 100U (CoC No. 1014) was again utilized to capture the primary stresses.

- The divider shell, CEC baseplate and MPC pedestal, were assumed to behave linear elastically to maximize impact loads.

- The VVM steel components were modeled with nonlinear elastic-plastic true stress-strain relationships

- The key input data of the VVM model is listed in FSAR Table 3.1.13.

3.2.3.2 Multi-Purpose Canister (MPC)

As stated previously, only relevant information necessary to evaluate the interaction between the HI-STORM UMAX VVM and the MPCs was presented in the HI-STORM UMAX Canister Storage System application. The applicant provided a finite element model of the MPC in order to demonstrate that any loads imparted on the MPC are within the licensing basis previously approved in Coc No. 1032 and to demonstrate that impact of the MPC into the radial guides of the CEC are within the limits of the materials of construction.

Key features of the applicant's FEA model included:

- The contents (fuel assemblies, fuel basket, and basket shims) of the MPC are explicitly modeled to account for the interaction between the MPC shell and the MPC contents.

- A refined finite element mesh at areas of interest in the canister was used to accurately capture the primary membrane and bending stresses as well as secondary stresses.

- The MPC shell and fuel basket were modeled using LS-DYNA thick shell elements.

- MPC lid, baseplate and each fuel assembly were modeled using solid elements.

- The MPC lid weld was explicitly modeled using solid elements.

- The material properties of the MPC components were based on the bounding temperatures under normal storage condition.

- The fuel assembly model, was assumed to be linear elastic

- All other MPC structural members we modeled with their true stress-strain relationships

The key input data of the MPC enclosure vessel model is listed in FSAR Table 3.1.15.

3.2.3.3 HI-TRAC Transfer Cask

The applicant provided no additional analysis on the transfer cask beyond the qualifying analyses presented in the HI-STORM FW Cask System application (CoC No. 1032). The applicant is not seeking (nor requires) additional approval to use the transfer cask as part of the HI-STORM UMAX Canister System since it has already been evaluated and approved by the staff in CoC No. 1032

3.3 Weights and Centers of Gravity

FSAR Table 3.2.1 contains bounding weight data of all types of MPCs, HI-TRACs and transporters that may be utilized with the HI-STORM UMAX Canister Storage System.

FSAR Table 3.2.2 contains bounding dimensional data for the MPC types certified for use in the HI-STORM FW Cask System.

3.4 Structural Analysis

The structural analysis for the HI-STORM UMAX Canister Storage System is presented in the FSAR Chapter 3. According to the FSAR, the HI-STORM UMAX Canister Storage System components are designed to protect the cask contents from significant structural degradation, provide adequate shielding, and maintain subcriticality and confinement under the design basis normal, off-normal, and accident loads. Individual loads for the three design conditions of normal, off-normal and accident conditions, including natural phenomena, have been addressed in FSAR Sections 2.3.2, 2.3.3 and 2.3.4.

3.4.1 Normal Conditions

As described in the FSAR, the HI-STORM UMAX Canister Storage System is designed to withstand normal conditions of storage, which include dead weight, handling (lifting of loaded MPC, lifting and handling of HI-TRAC VW with loaded MPC, lifting and transfer to ISFSI VVM with loaded MPC), pressure, temperatures, and snow and ice.

3.4.2 Off-Normal Conditions

The FSAR indicates that the HI-STORM UMAX Canister Storage System is designed to withstand off-normal conditions, which include pressure, environmental temperatures, transient event temperatures, leakage of seals, and partial blockage of air inlets.

3.4.3 Accident Conditions

3.4.3.1 Non-Mechanistic Tipover

The MPCs were evaluated by the applicant for a non-mechanistic tipover in CoC No. 1032. The staff found the applicant's evaluation that non-mechanistic tipover is not a credible event acceptable in that evaluation. No additional evaluation was necessary for the UMAX because UMAX is an underground storage system rendering non-mechanistic tipover not possible.

3.4.3.2 Dead load plus design basis explosion pressure on VVM components

The applicant evaluated the VVM Closure Lid and CEC shell utilizing strength of materials calculations. The applicant demonstrated through calculations that the dead load and explosion pressure with material property values at 640 F are bounded by the maximum vertical missile impact force by a factor of 3 and that the bearing capacity of the engineered fill (552 psi) exceeds the loading imparted by the design basis explosion.

3.4.3.3 Maximum Temperature and Internal Pressure Under Accident Conditions

The applicant determined that the configuration of the UMAX VVM does not allow internal pressure gradients since it is open to the environment. Maximum temperatures and pressures for the MPC-37 and MPC-89 are bounded by the evaluations presented in CoC No.-1032 for the same MPC designs.

3.4.3.4 Design Basis Fire on VVM Closure Lid

The applicant determined that the stresses in the closure lid due to a 10 psi explosion overpressure with material properties consistent with a 640 F temperature were extremely small. Given that a typical 30 minute hydrocarbon fire event occurs at 800 F and the design features of the ISFSI pad for rainwater diversion keep the fire at a minimum standoff distance, the applicant concluded that reduced material properties due to elevated temperature are not sufficient to exceed material allowables leading to a reduction in effectiveness of the closure lid. In addition, since the vertical large missile impact bounds the 10 psi explosion overpressure, the applicant also concluded that the design basis fire is also bounded by explosive overpressure.

3.4.3.5 Design Basis Flood

The applicant determined that sliding or other displacement of the stored contents due to moving water is not a credible event due to the specific design features of the UMAX Canister Storage System as an underground storage system.

3.4.3.6 Design Basis Missile Loading

3.4.3.6.1 Tornado Missile Strike on VVM Closure Lid

Due to the design features of the UMAX Canister Storage System as an underground storage system, the applicant determined that a missile strike is only credible for a side or top impact of the VVM closure lid. The applicant's strength of materials analysis of a design basis missile

strike on the closure lid demonstrated that the lid remained in place and did not collapse. The applicant's result determined that the shielding effectiveness of the closure lid was not reduced.

3.4.3.6.2 Tornado Missile Protection during Construction

The applicant's strength of materials analysis of a design basis missile strike on the exposed engineered fill demonstrated that the CEC and MPC are unaffected. FSAR Table 3.4.8 summarizes the missile impact analysis results and associated safety factors.

3.4.3.7 Design Basis Earthquake

A. Design Basis Seismic Model and SSI Analysis

The applicant's dynamic simulation of the structural response of the buried VVM was performed using the commercial finite element code, LS-DYNA. The seismic input for the transient finite element (FE) SSI analysis was an acceleration-time history set developed using Regulatory Guide (R.G.) 1.60 response spectra. The applicant determined that the acceleration time histories met the bounding spectra and power spectral density requirements of the "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition," NUREG-0800, section 3.7.1. The design basis earthquake (DBE) specified a horizontal zero period acceleration (ZPA) of 1.0 g and a vertical ZPA of 0.75g at the ground surface and a horizontal zero period acceleration (ZPA) of 0.93 g and a vertical ZPA of 0.71g at the foundation surface pad per FSAR Table 2.3.2.

The applicant's soil structure model development consisted of a two-step process utilizing SHAKE2000 and LS-DYNA to generate the response spectra at various ISFSI elevations with lower bound soil properties which were intended to bound the soil conditions at most US nuclear power plants. The SHAKE2000 analysis was performed first to generate the average strain compatible shear wave velocities as well as to extract the acceleration time history at the base of the soil column which is subsequently used in the LS-DYNA seismic response analysis (no structure present) and the LS-DYNA SSI analysis. This model also formed the basis for comparison of the site specific seismic and SSI analyses to determine whether site conditions are bounded by the general provisions set forth in the proposed CoC. The analytical approach described above was identical to that used for the previously approved amendments 7 and 9 to CoC No. 1014. FSAR Table 3.4.3 lists the peak ISFSI interface loads obtained from the LS-DYNA SSI simulations of the following four loading scenarios:

Scenario 1: All storage locations loaded with maximum weight MPCs, and a loaded VCT is placed at the center of the ISFSI.

Scenario 2: Same as Scenario 1 except that the Young's Modulus of the SFP concrete is reduced to one-half of its nominal value.

Scenario 3: Same as Scenario 1 except that the subgrade adjacent to one side of SES (Space A) is excavated down to the SFP and that the VCT is not considered.

Scenario 4: Same as Scenario 3 except that the Young's Modulus of the SFP concrete is reduced to one-half of its nominal value.

The applicant used the peak interface loads in the structural qualification of the VVM components and the ISFSI structures.

B. Seismic Qualification of VVM Components

In a seismic event, the loaded MPC in the HI-STORM UMAX VVM could experience impact loading from the MPC guides attached to the divider shell of the VVM.

The MPC enclosure vessel and contents were modeled by the applicant explicitly to correspond to the modeling techniques utilized for the MPC shell in CoC No. 1014, Amendment No. 9 (approved by the staff), to capture the high stress gradient at the impact location. The combination of results obtained from the MPC impact analysis and those from the SSI analyses were used to structurally qualify HI-STORM VVM components. FSAR Table 3.4.4 summarizes the seismic qualification analysis results for VVM components.

C. Strength Qualification of the ISFSI Structure

The applicant evaluated the strength qualification of the ISFSI structures under design basis seismic loading by extracting the peak interface loads obtained from the SSI analyses and applying them to a quasi-static finite element analysis. The applicant used the actual input loads which were larger than the peak loads obtained from the LS-DYNA analyses to provide an additional margin of safety. The applicant determined the SFP, TSP, and enclosure wall met the American Concrete Institute (ACI)-318 (2005) strength limits for all load combinations applicable for this design. The quasi-static structural analysis utilized the ANSYS finite element analysis software. The following is a summary of the applicant's model formulation:

- SFP, TSP, Subgrade beneath TSP modeled with elastic SOLID45
- The lateral subgrade adjacent to the ISFSI is included in the FE model
- The element mesh is appropriately refined in areas of load application on the SFP and the TSP.
- Quarter symmetry is utilized
- Simulation Model II, as described below, uses a full FE model since it is non-symmetric

The following is a summary of the VVM loading configurations considered:

Simulation Model I: all the storage locations in the ISFSI are populated and experience identical bounding peak vertical seismic loading

Simulation Model II: two rows of VVM locations adjacent to the symmetry line loaded

Simulation Model III*:* single middle row of VVM is loaded

Simulation Model IV: single VVM loaded centered near the periphery of the ISFSI

Simulation Model V: similar to Model I but with lateral subgrade surrounding the retaining walls removed. Effects of transporter also not considered since loading activities will be suspended during excavations.

Simulation Models I, II, III and IV, applied the peak bearing load from the LS-DYNA SSI analysis from a single transporter track as a static load to both transporter tracks footprints simultaneously. The applicant took no credit for the dynamic increase factor of 25% for flexure and 10% for shear permitted by in the strength qualification of reinforced concrete in order to provide additional conservatism in the analysis.

Table 3.4.5 of the FSAR illustrates that the ISFSI pad and support foundation pad strength has significant margin over ACI 318 allowable stresses.

The applicant noted that the structural analysis of the ISFSI considered the peak dynamic loads (unfiltered) from the LS-DYNA SSI analysis and that it is permissible to use equivalent static loads obtained by removing high frequency components using appropriate filters. The applicant, however, failed to provide a definition of an “appropriate filter” or information justifying the use of filters. Since no information was presented by the applicant the NRC did not accept the use of filters to establish loadings for the HI-STORM UMAX Canister Storage System in performing its evaluation.

3.5 Staff Evaluation

The staff evaluation of the licensee’s structural models and other calculations to support the structural analyses included review of the engineering drawings to verify that adequate geometry dimensions were translated to the analysis models, review of the material properties presented in the FSAR relevant to structural performance to verify that they were used appropriately and properly referenced, confirmation of finite element input values used in the licensee calculation packages, along with a review of design details used to provide parameters in the computer models and other calculations.

The staff determined that the proper material properties and boundary conditions were used based on well-established structural engineering methods and analytical techniques. The staff determined that the licensee’s selected analytical models, assumptions, and other calculations accurately reflected the specific design parameters, and that the assumptions and modeling parameters were consistent with review guidelines in NUREG-1536 and ISG-21, “Use of Computational Software” as well as 10 CFR 72.122. The staff determined that licensee assumptions were adequate for the structural performance characteristics in the HI-STORM UMAX Canister Storage System geometry and analyzed conditions. Finally, the staff determined that the licensee-provided FSAR sections included accurate information that allowed the staff to make a safety determination on the acceptability of the proposed design.

Therefore the staff finds that the HI-STORM UMAX Canister Storage System structural analysis and conclusions are acceptable and that the HI-STORM UMAX Canister Storage System will safely store spent nuclear fuel within TS parameters.

3.6 Evaluation Findings

F3.1 The FSAR adequately describes all SSCs that are ITS, providing drawings and text in sufficient detail to allow evaluation of their structural performance.

F3.2 The applicant has met the requirements of 10 CFR Part 72.236(b). The SSCs ITS are designed to accommodate the combined loads of normal or off-normal operating conditions and accidents or natural phenomena events with an adequate margin of safety. Stresses at various locations of the cask for various design loads are determined by analysis. Total stresses for the combined loads of normal, off-normal, accident, and natural phenomena events are acceptable and are found to be within limits of applicable codes, standards, and specifications.

F3.3 The applicant has met the requirements of 10 CFR Part 72.236(c), for maintaining subcritical conditions. The structural design and fabrication of the DSS includes structural margins of safety for those SSCs important to nuclear criticality safety. The applicant has demonstrated adequate structural safety for the handling, packaging, transfer, and storage under normal, off-normal, and accident conditions.

F3.4 The applicant has met the requirements of 10 CFR 72.236(l), "Specific Requirements for Spent Fuel Storage Cask Approval." The design analysis and submitted bases for evaluation acceptably demonstrate that the cask and other systems important to safety will reasonably maintain confinement of radioactive material under normal, off-normal, and credible accident conditions.

F3.5 The applicant has met the requirements of 10 CFR 72.236 with regard to inclusion of the following provisions in the structural design:

- Design, fabrication, erection, and testing to acceptable quality standards.

- Adequate structural protection against environmental conditions and natural phenomena, fires, and explosions.

- Appropriate inspection, maintenance, and testing.

- Adequate accessibility in emergencies.

- A confinement barrier that acceptably protects the cladding during storage.

- Structures that are compatible with appropriate monitoring systems.

- Structural designs that are compatible with retrievability of SNF.

F3.6 The applicant has met the specific requirements of 10 CFR 72.236(g) and as they apply to the structural design for spent fuel storage cask approval. The cask system structural design acceptably provides for the following required provisions:

- Storage of the spent fuel for a minimum required years.

- Compatibility with wet or dry loading and unloading facilities.

The staff concludes that the structural properties of the SSCs of the HISTORM UMAX Canister Storage System are in compliance with 10 CFR Part 72, and that the applicable design and acceptance criteria have been satisfied. The evaluation of the structural properties provides reasonable assurance that the HISTORM UMAX Canister Storage System will allow safe storage of SNF for a licensed (certified) life of 20 years. This finding is reached on the basis of a review that considered the regulation itself, appropriate regulatory guides, applicable codes and standards, and accepted engineering practices.

4 THERMAL EVALUATION

The staff's thermal review ensures that the cask components and fuel material temperatures of the HI-STORM UMAX Canister Storage System will remain within the allowable values under normal, off-normal, and accident conditions. These objectives include confirmation that the fuel cladding temperature will be maintained below specified limits throughout the storage period to protect the cladding against degradation that could lead to gross ruptures. This portion of the review also confirms that the cask thermal design has been evaluated using acceptable analytical techniques and/or testing methods. The review was conducted to the appropriate regulations as described in 10 CFR 72.236 that identify the specific requirements for spent fuel storage cask approval and fabrication. The unique characteristics of the spent fuel to be stored are identified, as required by 10 CFR 72.236(a), so that the design basis and the design criteria that must be provided for the SSCs ITS can be assessed under the requirements of 10 CFR 72.236(b). This application was also reviewed to determine whether the HI-STORM UMAX Canister Storage System design fulfills the acceptance criteria listed in Sections 2, 4 and 12 of NUREG-1536, Revision 1, as well as applicable interim staff guidance (ISG).

4.1 Spent Fuel Cladding

The applicant adopted certain guidelines of NRC, "Standard Review Plan for Dry Cask Storage Systems," NUREG-1536, Revision No. 1 and ISG-11, Revision 3, "Cladding Considerations for the Transportation and Storage of Spent Fuel", to demonstrate the safe storage of the material content described in FSAR Chapter 2 and in the CoC application for those aspects relevant to the HI-STORM UMAX Canister Storage System design. As explained later in this SER section, the staff has determined that the applicant demonstrated the HI-STORM UMAX Canister Storage System complies with the following requirements:

1. The fuel cladding temperature must meet the temperature limit appropriate to its burnup level and condition of storage or handling set forth in FSAR Table 2.3.7.

2. The maximum internal pressure of the MPC should remain within its design pressures for normal, off-normal, and accident conditions set forth in FSAR Table 2.3.5.

3. The temperatures of the cask materials shall remain below their recommended limits set forth in FSAR Table 2.3.7.

4.2 Thermal Properties of Materials

Material property tables for the HI-STORM UMAX components are included in FSAR Section 4.2. Materials present in the MPCs include Alloy X (defined in the FSAR), Metamic-HT, aluminum, and helium. Materials present in the HI-STORM UMAX Canister Storage System underground storage vertical ventilated module (VVM) include carbon steel, concrete, insulation, and ambient air. Thermal properties provided in the FSAR include thermal conductivity, density, heat capacity, gas viscosity, and emissivity. The temperature range for the material properties covers the range of temperatures encountered during the thermal analysis.

The staff evaluated the applicant's thermal properties used to perform the thermal evaluation of the HI-STORM UMAX Canister Storage System. Based on the information provided in the application regarding thermal properties, the staff determined that the application is consistent with guidance provided in Section 4.5.4.2 (Material Properties) of NUREG-1536 that states the applicant should include thermal properties for all components used in the calculational model. The staff determined that the thermal properties used in the safety analysis are appropriate because they cover the temperature range encountered during normal, off normal, and accident conditions. Therefore, the staff concludes that the thermal properties have been appropriately identified and are acceptable because the properties satisfy NUREG-1536 and the requirements in in 10 CFR 72.122(h)(1), 72.122(l), 72.236(b), 72.236(f), 72.236(g), and 72.236(h).

4.3 Specifications for Components

HI-STORM UMAX Canister Storage System materials and components designated as ITS (i.e., required to be maintained within their safe operating temperature ranges to ensure their intended function) are summarized in FSAR Tables 2.0.1 through 2.0.6. For evaluation of HI-STORM UMAX Canister Storage System thermal performance, material temperature limits for long term normal, short-term operations, and off-normal and accident conditions are provided in FSAR Table 2.3.7. Fuel cladding temperature limits included in FSAR Table 2.3.7 are adopted from ISG-11. These limits are applicable to all fuel types, burnup levels, and cladding materials approved by the NRC for power generation.

The staff reviewed the applicant's specifications for ITS SSCs for the HI-STORM UMAX Canister Storage System. Based on the information provided in the application regarding specifications for components, the staff determined that the application is consistent with guidance provided in Section 4.4.2 (Material and Design Limits) of NUREG-1536 that provides that cask components and fuel materials should be maintained between their minimum and maximum temperature limits for normal, loading, off-normal, and accident-level conditions to enable all components to perform their intended safety function. Therefore, the staff concludes that the specifications for components are acceptable because the material temperature limits satisfy NUREG-1536 and the requirements in in 10 CFR 72.236(b), 72.236(f), 72.236(g), and 72.236(h).

4.4 HI-STORM UMAX Canister Storage System

4.4.1 General Description

As described in FSAR Section 1.2, the HI-STORM UMAX Canister Storage System consists of interchangeable MPCs, which maintain the configuration of the fuel and operate as a confinement boundary between the stored spent nuclear fuel and the environment; and a storage module that provides structural protection and radiation shielding during long-term

storage of the MPC. The HI-STORM UMAX VVM provides for storage of the MPC in a vertical configuration inside a subterranean cylindrical cavity entirely below the top-of-grade of an independent spent fuel storage installation. The key constituents of a HI-STORM UMAX VVM are:

1) The Cavity Enclosure Container (CEC)
2) The Closure Lid
3) The ISFSI Pad
4) The Support Foundation Pad
5) The Subgrade and Under-grade
6) The Enclosure Walls (optional)

A detailed description of MPCs and the HI-TRAC transfer cask are provided in Section 1.2 of the HI-STORM UMAX FSAR.

4.4.2 Design Criteria

The applicant's thermal design and operation of the MPC in the HI-STORM UMAX system follows the review guidance in NUREG-1536 and ISG-11, Revision 3. Specifically, provisions from guidance that are invoked by the applicant are:

1. The thermal acceptance criteria for all commercial spent fuel (CSF) authorized by the USNRC for operation in a commercial reactor are unified into one set of requirements.

2. The maximum value of the calculated temperature for all CSF under long-term normal conditions of storage must remain below 400ºC (752ºF). For short-term operations, including canister drying, helium backfill, and on-site cask transport operations, the fuel cladding temperature must not exceed 400ºC (752ºF) for high burnup fuel (HBF) and 570ºC (1058ºF) for moderate burnup fuel.

3. The maximum fuel cladding temperature as a result of an off-normal or accident event must not exceed 570ºC (1058ºF).

4. For HBF, operating restrictions are imposed to limit the maximum temperature excursion during short-term operations to 65ºC (117ºF) and the number of excursions to less than 10.

As described in FSAR Section 2.0.3, the HI-STORM UMAX VVM rejects heat from the stored MPCs by delivering cool ambient air to the annular space around the MPC. The ambient air undergoes progressive heating and reduction in density as it rises in the cylindrical space surrounding the MPC through convective heat transfer with the MPC shell, and exits the cell through the vertical flue mounted on the central region of the closure lid. The storage cavities have a constant out flow of air which will tend to retard the deposition of air borne particulates and debris in the storage space. The accumulated solids can be vacuumed out of the storage cavity by standard means. As shown in FSAR Chapter 4, the VVMs are designed to reject the maximum allowable heat load. The VVM is designed for extreme cold conditions.

4.4.3 Design Features

According to the FSAR, the HI-STORM UMAX Canister Storage System is designed, with multiple cooling passages and suitably sized flow annuli, which maximize air flow by ensuring a turbulent flow regime at design basis heat loads. Cooling air to each MPC storage cavity is provided by four independent ducts. Thus, there is a significant level of redundancy in the cooling air delivery system for the HI-STORM UMAX Canister Storage System. The air inlet locations are separated from the outlet vent by a significant lateral and vertical distance. This design feature ensures that there is minimal mixing of cold and hot air in the storage system. Calculations summarized in FSAR Chapter 4 show that the heat rejection performance of the system is stable under varying wind speed.

To ensure the permissible PCT limits are not exceeded, FSAR Subsection 2.1.9 specifies the maximum allowable decay heat per assembly for each MPC model in the different thermal patters. FSAR Tables 2.1.8 and 2.1.9 summarize the heat load data for MPC-37 and MPC-89.

The staff reviewed the applicant's general description, design criteria, and design features of the HI-STORM UMAX storage system. Based on the information provided in the application regarding these items, the staff determines that the application is consistent with guidance provided in Section 4.4.1 (Decay Heat Removal System) of NUREG-1536 which provides that the applicant should present a detailed description of the proposed cask heat removal system and its passive cooling characteristics. Here, the applicant has provided a detailed description of decay heat removal system and its passive cooling characteristics. Therefore, the staff concludes that the description of the decay heat removal system is acceptable because the description satisfies NUREG-1536 and the requirements in in 10 CFR 72.122(h)(1), 72.122(l), 72.236(b), 72.236(f), 72.236(g), and 72.236(h).

4.5 Thermal Model

The applicant used FLUENT program to evaluate the thermal performance of the HI-STORM UMAX Canister Storage System. FLUENT is a finite volume computational fluid dynamics (CFD) program with capabilities to predict fluid flow and heat transfer phenomena in two and three dimensions. The three-dimensional (3-D) thermal analysis model for the VVM developed by the applicant is described below. The MPC and basket thermal models are described in the HI-STORM FW FSAR.

The airflow through the cooling passages of the VVM is modeled as turbulent, using the k-ω model with transitional option enabled. The underside of the SFP is assumed to be supported on a subgrade at an isothermal surface temperature. A quarter-symmetry model for the VVM assembly seeks to represent the essential geometry details of the physical system as depicted in the Licensing Drawings in FSAR Section 1.5 with the assumptions as summarized below:

1. In FSAR Table 2.1.7, the fuel assemblies loaded in MPC-37 are catalogued as short, standard and long fuel. For each length catalogued, the minimum active fuel length is used in the model. For instance, the active fuel lengths of 128", 144" and 168" are used to model the short, standard and long fuel respectively. This is conservative, because the shorter active fuel length has the higher heat load density, which results in a higher peak cladding temperature (PCT).

2. The soil Subgrade beneath the VVM assembly is assumed to be equal to the design basis soil temperature in FSAR Table 2.3.6.

3. The side surface of VVM is assumed to be insulated.

The applicant stated that that the axial variation of the heat generation rate in the design basis fuel assembly is defined based on the axial burnup distribution and that this distribution is used for analyses only, and does not provide a criteria for fuel assembly acceptability for storage in the HI-STORM UMAX Canister Storage System. Because different distributions may negatively affect the FSAR thermal results and new thermal analysis may be required to demonstrate that thermal limits are not exceeded, the applicant performed the analysis using two axial heat load distributions to address this issue:

Case1 (flattened axial distribution as depicted in the licensing basis profile in Table 2.1.5 and Figure 2.1.3 of the FSAR) and

Case 2 (axial distribution with peak to average heat generation rate equal to 1.2 in every storage cell.

Based on the analysis results the applicant determined a small difference between the two axial profiles. In order to ensure the assumed axial variation would bound other axial distributions during vacuum drying, the applicant performed additional analysis using a center biased heat generation profile. The applicant determined that in order to comply with the permissible peak cladding temperature limit, a reduction in the threshold heat load allowed for vacuum drying was needed, as specified in the FSAR.

The staff reviewed the applicant’s description of the HI-STORM UMAX Canister Storage System thermal model. Based on the information provided in the application regarding the thermal model, the staff determines that the application is consistent with guidance provided in Section 4.4.4 (Analytical Methods, Models, and Calculations) of NUREG-1536 that provides that the applicant should present a thermal analysis that clearly demonstrates the storage system’s ability to manage design heat loads and have the various materials and components remain within temperature limits. Here, the applicant has provided a detailed description of thermal models used to perform the evaluation of the storage cask and the results of the model, as confirmed by staff, demonstrate the storage system’s ability to manage design heat loads and have the materials and components remain within temperature limits. Therefore, the staff concludes that the description of the thermal model is acceptable because the description satisfies NUREG-1536 and the requirements in in 10 CFR 72.122(h)(1), 72.122(l), 72.236(b), 72.236(f), 72.236(g), and 72.236(h).

4.6 Thermal Evaluation for Normal Conditions of Storage

The applicant used the 3-D model described in the previous section to determine temperature distributions under long-term normal storage conditions for both MPC-89 and MPC-37. FSAR Tables 4.4.2, 4.4.7, 4.4.9, and 4.4.10 provide key thermal and pressure results. Based upon The applicant’s results, the temperature field in the HI-STORM UMAX Canister Storage System with a loaded MPC containing heat emitting spent nuclear fuel complies with all regulatory temperature limits (FSAR Table 2.3.7). The staff confirmed the thermal environment in the HI-STORM UMAX Canister Storage System is in compliance with FSAR Chapter 2 Design Criteria. As explained in FSAR Chapter 3 and evaluated in Chapter 3 of this SER, all HI-STORM UMAX VVM and MPC materials of construction will satisfactorily perform their intended function in the storage mode under a minimum temperature condition of -40°F.

The storage scenarios described above evaluated the HI-STORM UMAX Canister Storage System up to an elevation of 1500 feet. However, if an ISFSI is located at an elevation greater than 1500 ft, the effect of altitude on the peak cladding temperature shall be quantified as part of the 10 CFR 72.212 evaluations for the site using the site ambient conditions.

The applicant calculated the MPC maximum gas pressure for a postulated release of fission product gases from fuel rods into the MPC free space. For these scenarios, the amounts of each of the release gas constituents in the MPC cavity are summed and the resulting total pressures determined from the ideal gas law. Based on fission gases release fractions (NUREG 1536 criteria), rods' net free volume and initial fill gas pressure, maximum gas pressures with 1% (normal), 10% (off-normal) and 100% (accident condition) rod rupture are given in FSAR Table 4.4.7. The maximum computed gas pressures reported in FSAR Table 4.4.7 are all below the MPC internal design pressures for normal, off-normal and accident conditions specified in FSAR Table 2.3.5.

The applicant performed grid sensitivity studies to obtain the discretization error. For this purpose, the applicant generated five different grids. Using the results from these grids, the applicant calculated the grid convergence index (GCI). Because the results of the GCI calculation were an apparent order larger than the analytical one, the applicant recalculated the GCI using a set of three grids. This calculation showed an apparent order smaller than two. However, as indicated in American Society of Mechanical Engineers (ASME), "Standard for Verification and Validation in Computational Fluid Dynamics and Heat Transfer," V&V 20-2009, in order to demonstrate that the apparent order is constant for a series of simulations, a minimum of four grids is required. Per V&V 20-2009, three-grid solution for the observed order p may be adequate if some of the values of the variable of interest (for example, peak cladding temperature) predicted on the three grids are in the asymptotic region for the simulation series. The staff determined the GCI calculation provided by the applicant did not demonstrate that the apparent order was constant nor did it show that the results were in the asymptotic region. The staff requested the applicant's uncertainty of the analytical results associated with the boundary conditions (e.g., heat transfer coefficient), fuel effective thermal conductivity, and porous media flow resistance factors (used to model the fuel assemblies). The applicant performed additional analyses to obtain the sensitivity of the analytical results to the fuel effective thermal conductivity and porous media flow resistance factors. The calculations included a 10% reduction in the effective thermal conductivity and a porous media flow resistance factor of about one million. The applicant's calculated friction factor used in the PWR fuel assembly porous media model was found to be non-conservative because the staff calculated a higher friction factor based on thermal-hydraulic characterization performed by Sandia National Laboratory (SNL). Staff's thermal analysis of SNL's thermal-hydraulic experiment data, indicated that a fuel assembly viscous resistance factor of about a million would match the thermal-hydraulic experimental data obtained by SNL. Therefore, in order to ensure that the fuel, MPC, and cask components remain below their respective temperature limits, the applicant lowered the allowable heat loads originally requested, as indicated in the FSAR.

For wind studies, the applicant developed a separate model to obtain the effect of wind and the effect of air mixing. Although, based on observations and measurements (from National Oceanic and Atmospheric Administration data), low speed wind is a normal environmental variable, the applicant stated that wind should be treated as an off-normal occurrence, but provided no additional justification. Per ANSI/ANS-57.9, off-normal operations are those conditions which, although not occurring regularly, are expected to occur no more than once a year. Low wind speed (e.g. 7 mph wind) in any given direction occurs much more frequently,

and, therefore, should be treated as a normal condition. Treating low speed wind as a normal occurrence requires the use of the long term storage recommended temperature limit (752°F or 400°C.) Therefore, the staff concluded wind should be treated as a normal environmental variable and should be included in the analysis, especially since it has an effect on the predicted peak cladding temperature as compared to the quiescent conditions.

The applicant explained that the effect of wind in the inlet temperature is included in the analysis by adding the increase in the inlet temperature (obtained from analysis which considers a one by eight array model, as described in the FSAR) to the peak cladding temperature predicted by the thermal model developed for wind studies. In adding the two, the applicant predicted a PCT that was below, but very close to the ISG-11 recommended limit.

Chapter 4 (Thermal Evaluation) of NUREG-1536 provides the need to assess modeling details such as simulation options, simplifications, and accuracy of results. It also states that for any computational modeling software to demonstrate that a particular cask design satisfies regulatory requirements, adequate validation of that computational modeling software must be demonstrated by the applicant. As defined in NUREG-1536, validation is a demonstration of the validity of a computer code for use in a general area of application by comparison of the code's calculational results with the measured results (data) from a variety of experiments spanning the area of intended applications. The applicant performed the analysis but the application did not include any validation data to determine the accuracy of the analytical results. Therefore, without adequate validation of the analytical models, the staff was unable to determine the accuracy of the calculations and claimed thermal margins. Although the applicant proposed that a HI-STORM UMAX thermal test would be added to the CoC to address the staff's concern, the staff found the applicant's approach unacceptable because the proposed approach postpones the validation of the analytical methods after the CoC is issued. Therefore, in order to compensate for the lack of validation of the thermal models and uncertainty of the calculations, the applicant reduced the total heat load by 20%. The staff performed additional calculations to determine the additional margin that would exist with a 20% reduction in the total heat load. The staff's results were consistent with the FSAR, which includes a calculation at 80% of design basis heat load. Based on these calculations the staff determined that the 20% reduction would provide sufficient margin against the recommended peak cladding temperature limit. The staff determined that this margin against the recommended limit was sufficient to provide reasonable assurance because it would compensate for uncertainties associated with the modeling and application errors.

The staff reviewed the applicant's thermal evaluation of the HI-STORM UMAX Canister Storage System during normal conditions of storage. Based on the information provided in the application regarding the thermal model and evaluation, the staff determines that the application is consistent with guidance in Section 4.4.4 (Analytical Methods, Models, and Calculations) of NUREG-1536, that the applicant should present a thermal analysis that clearly demonstrates the storage system's ability to manage design heat loads and have the various materials and components remain within temperature limits. Here, the applicant has provided a thermal evaluation used to show that calculated maximum temperatures remain below the recommended limits described in the application. However, the applicant did not provide evidence that analytical models have been validated, as required by NUREG-1536. To compensate for the lack of validation and uncertainty determination, the applicant reduced the total heat load by 20%. The staff determined that a 20% reduction in the total heat load would be sufficient to provide additional margin against the recommended cladding temperature limit.

The staff concluded that this additional margin would compensate for the uncertainty in the applicant's thermal models.

4.7 Thermal Evaluation for Short-Term Operations

The applicant evaluated the vacuum drying condition and normal onsite transfer (FSAR Section 4.5). The applicant incorporated by reference other short-term operations involving the HI-TRAC VW transfer cask.

4.8 Off-Normal and Accident Events

4.8.1 Off-Normal Events

The applicant considered five off-normal events: off-normal pressure, off-normal environmental temperature, partial blockage of air inlets, off-normal malfunction of forced helium dehydrator (incorporated by reference to HI-STORM FW, and sustained wind. (As discussed in section 4.6, the review of the applicant's off-normal events did not include a review of sustained winds because sustained winds from 0 to 10 miles per hour (mph) are normal occurrence and should apply the normal temperature limit.) The MPC off-normal pressures are reported in FSAR Table 4.6.5. The results are below the off-normal design pressure (FSAR Table 2.3.5). The off-normal temperature results are provided in FSAR Table 4.6.1. The results are below the off-normal condition temperature (FSAR Tables 2.3.7). The computed temperatures for the partial blockage of air inlets are reported in FSAR Table 4.6.1 and the corresponding MPC internal pressure in FSAR Table 4.6.5. The results are confirmed to be below the temperature and pressure limits (FSAR Table 2.3.7 and 2.3.5) for off-normal conditions.

The staff reviewed the applicant's thermal evaluation during off-normal conditions and verified the maximum cladding temperatures predicted by the applicant would remain below ISG-11 Rev. 3 recommended limit of 570°C for all postulated off-normal events. Based on the information provided in the application regarding off-normal events, the staff determines that the application is consistent with guidance provided in Section 4.4.4 (Analytical Methods, Models, and Calculations) of NUREG-1536 which provides that the applicant should present a thermal analysis that clearly demonstrates the storage system's ability to manage design heat loads and have the various materials and components remain within temperature limits. Here, the applicant has demonstrated this ability by performing the calculations and demonstrating that the analysis results are lower than the recommended limit of 570°C. Therefore, the staff concludes that the thermal evaluation during off-normal events is acceptable because the thermal evaluation satisfies NUREG-1536 and the requirements in in 10 CFR 72.122(h)(1), 72.122(I), 72.236(b), 72.236(f), 72.236(g), and 72.236(h).

4.8.2 Accident Events

The applicant considered six accident events: fire, jacket water loss, extreme environmental temperatures, 100% blockage of air ducts, burial under debris, and flood. Accident analyses results are provided in FSAR Tables 4.6.6, 4.6.7, 4.6.9, and 4.6.10. All predicted maximum temperatures and pressures remain below the accident limits defined in FSAR Table 2.3.5 (accident design pressure) and Table 2.3.7 (accident temperature limit).

The staff reviewed the applicant's thermal evaluation during accident conditions and verified the maximum cladding temperatures predicted by the applicant would remain below ISG-11 Rev. 3

recommended limit of 570°C for all postulated accident events. Based on the information provided in the application regarding accident events, the staff determines that the application is consistent with guidance provided in NUREG-1536, Section 4.4.4 (Analytical Methods, Models, and Calculations) that provides that the applicant should present a thermal analysis that clearly demonstrates the storage system's ability to manage design heat loads and have the various materials and components remain within temperature limits. Here, the applicant has demonstrated this ability by performing the calculations and demonstrating that the analysis results are lower than the recommended limit of 570°C. Therefore, the staff concludes that the thermal evaluation during accident events is acceptable because the thermal evaluation satisfies NUREG-1536 and the requirements in in 10 CFR 72.122(h)(1), 72.122(l), 72.236(b), 72.236(f), 72.236(g), and 72.236(h).

4.9 Confirmatory Analysis

The staff reviewed the applicant's thermal models used in the analyses. The staff checked the code input in the calculation packages and confirmed that the proper material properties and boundary conditions were used. The staff verified that the applicant's selected code models and assumptions were adequate for the flow and heart transfer characteristics prevailing in the HI-STORM UMAX Canister Storage System geometry and analyzed conditions. The engineering drawings were also consulted to verify that adequate geometry dimensions were translated to the analysis models. The material properties presented in the FSAR were reviewed to verify that they were appropriately referenced and used. The staff assured that the applicant performed appropriate sensitivity analysis calculations to obtain mesh-independent results that would provide bounding predictions for all analyzed conditions during normal fuel transfer and accidents. Finally, through request for additional information (RAI) the staff made sure the applicant provided an FSAR that included complete and accurate information for the staff to make a safety determination on the adequacy of HI-STORM UMAX Canister Storage System thermal design.

To verify the applicant's 3-D thermal model used to analyze wind conditions (which is the bounding case), the staff modified the applicant's FLUENT model to extend the location of the boundary to make sure it does not affect the applicant's predicted results. Using the confirmatory analysis, the staff verified the applicant's developed model is adequate to represent wind conditions.

4.10 Evaluation Findings

F4.1 FSAR Chapter 2 describes SSCs important to safety to enable an evaluation of their thermal effectiveness. Cask SSCs important to safety remain within their operating temperature ranges.

F4.2 The HI-STORM UMAX Canister Storage System is designed with a heat-removal capability having verifiability and reliability consistent with its importance to safety. The cask is designed to provide adequate heat removal capacity without active cooling systems.

F4.3 The spent fuel cladding is protected against degradation leading to gross ruptures under long-term storage by maintaining cladding temperatures below 752°F (400°C). Protection of the cladding against degradation is expected to allow ready retrieval of spent fuel for further processing or disposal.

F4.4 The spent fuel cladding is protected against degradation leading to gross ruptures under off-normal and accident conditions by maintaining cladding temperatures below 1058°F (570°C). Protection of the cladding against degradation is expected to allow ready retrieval of spent fuel for further processing or disposal.

F4.5 The staff finds that the thermal design of the HI-STORM UMAX Canister Storage System is in compliance with 10 CFR Part 72 and that the applicable design and acceptance criteria have been satisfied. The evaluation of the thermal design provides reasonable assurance that the cask will allow safe storage of spent fuel. This finding is reached on the basis of a review that considered the regulation itself, appropriate regulatory guides, applicable codes and standards, and accepted engineering practices.

5 CONFINEMENT EVALUATION

In the confinement review, the NRC evaluates the confinement features and capabilities of the proposed cask system. In conducting this evaluation, the NRC staff seeks to ensure that radiological releases to the environment will be within the limits established by the regulations and that the spent fuel cladding and fuel assemblies will be sufficiently protected during storage against degradation that might otherwise lead to gross ruptures.

The HI-STORM Canister Storage System is similar in design to the HI-STORM 100U system approved by the NRC in CoC No. 1014, Amendment No. 7. The major differences between them are that the HI-STORM UMAX Vertical Ventilated Module (VVM) Cavity is larger in diameter than the HI-STORM 100 VVM, and that the UMAX closure lid features a modified outlet ventilation duct system. Given that the HI-STORM UMAX Canister Storage System confinement vessel is the MPC that is similar to that evaluated in CoC No. 1014, Amendment No. 7, and that there is only a minor change in confinement system design between the HI-STORM UMAX Canister Storage System and the HI-STORM 100U, the confinement review was focused on evaluating the effects of the differences in design.

5.1 Confinement System

The confinement boundary is defined by the MPC shell, MPC baseplate, MPC lid, port cover plates, closure ring, and associated welds (the HI-STORM UMAX VVM does not serve as a confinement function). There are no bolted closures or mechanical seals in the MPC confinement boundary. The details of the MPCs are described in the HI-STORM FW docket.

Helium Leak Testing

All the confinement components (including the confinement welds and the base metals) of the HI-STORM UMAX are required for helium leak testing except the lid-to-shell weld since the weld meets the criteria of ISG-18. The confinement boundary of the HI-STORM UMAX is helium leakage tested to be leak-tight (1.0×10^{-7} ref-cm^3/sec) in accordance with the leakage test methods and procedures of ANSI N14.5-1997.

5.2 Staff Evaluation

The staff reviewed the description of HI-STORM UMAX Canister Storage System confinement system and concludes that confinement of all radioactive materials in the HI-STORM UMAX system is provided by the MPC which remains unchanged from those used in the HI-STORM FW which was approved and licensed. The staff also concludes that; as in HI-STORM FW confinement components, the material of construction (austenitic stainless steel) for the HI-STORM UMAX confinement vessel is known from extensive industrial experience to have high integrity, high ductility and high fracture strength welds, and the MPC enclosure vessel welds provide a secure barrier against leakage. Finally, the staff concludes that all the confinement components (including the confinement welds and the base metals) of the HI-STORM UMAX are required to be helium leak tested to assure they are leaktight (except the lid-to-shell weld since the weld meets the criteria of ISG-18).

After reviewing the descriptions of the confinement system in FSAR and referencing to HI-STORM FW, the staff agrees that leakage from the confinement boundary is not credible.

5.3 Evaluation Findings

F5.1 Chapter 7 of the FSAR sufficiently describes confinement structures, systems, and components important to safety. The cask is leaktight and its quantity of radioactive nuclides released to the environment satisfies the regulatory requirements of 10 CFR 72.104(a) and 10 CFR 72.106(b).

F5.2 The HI-STORM UMAX Canister Storage System design adequately protects the spent fuel cladding against degradation that might lead to gross ruptures.

F5.3 The HI-STORM UMAX Canister Storage System provides redundant sealing of the confinement system. There are no bolted closures or mechanical seals in the MPC confinement boundary.

F5.4 The cask confinement system was evaluated to demonstrate that it will reasonably maintain confinement of radioactive material under normal, off-normal, and credible accident conditions.

F5.5 All the confinement components (including the confinement welds and the base metals) of the HI-STORM UMAX Canister Storage System are required for helium leak testing except the lid-to-shell weld since the weld meets the criteria of ISG-18.

6 SHIELDING AND RADIATION PROTECTION EVALUATION

The shielding and radiation protection review evaluates the ability of the proposed shielding features to provide adequate protection against direct radiation from the DSS contents. The shielding features should limit the dose to the operating staff and members of the public so that the dose remains within regulatory requirements during normal operating, off-normal, and design-basis accident (DBA) conditions. The review seeks to ensure that the shielding design is sufficient and reasonably capable of meeting the operational dose requirements of 10 CFR 72.104 and 72.106 in accordance with 10 CFR 72.236(d).

6.1 Introduction

The staff's review considered the acceptance criteria specified in Section 6 of NUREG-1536, "Standard Review Plan for Dry Cask Storage Systems." The staff's review was performed based on information provided in the HI-STORM UMAX Canister Storage System FSAR.

The HI-STORM stores a sealed canister containing spent nuclear fuel in an in-ground Vertical Ventilated Module (VVM). The HI-STORM UMAX VVM is the underground equivalent of the HI-STORM FW storage module certified in CoC No. 72-1032. The HI-STORM UMAX Canister Storage System is also similar to the HI-STORM 100U VVM approved in CoC No. 1014, Amendment No. 7. The main differences between the HI-STORM UMAX VVM and HI-STORM 100 VVM are that the HI-STORM UMAX VVM cavity is larger in diameter and has a modified outlet ventilation duct system for the closure lid

6.2 Design Criteria

The objective of shielding is to assure that radiation dose rates at key locations are as low as practical in order to maintain occupational doses to operating personnel as low as reasonably achievable (ALARA) and to meet the requirements of 10 CFR 72.104 (a) to a maximum of 25 mrem/year whole body, 75 mrem/year thyroid, and 25 mrem/year for other critical organs, including contributions from all nuclear fuel cycle operations for a real individual beyond the controlled area boundary and 10 CFR 72.106 for dose at the controlled area boundary

The applicant stated that the three locations that are of particular interest in the storage mode are:
1. Immediate vicinity of the cask
2. Restricted area boundary
3. Controlled area (site) boundary

Dose rates in the immediate vicinity of the loaded VVM are important in consideration of occupational exposure.

The applicant stated since these limits are dependent on plant operations as well as site specific conditions, the determination and comparison of ISFSI doses to this limit are necessarily site-specific. The determination of site-specific ISFSI dose rates at the site boundary and demonstration of compliance with regulatory limits is to be performed by the licensee in accordance with 10 CFR 72.212.

The storage module is designed to limit the calculated surface dose rates on the cask for all MPC designs. The storage module provides structural protection and radiation shielding during long-term storage of the MPC. In addition, the HI-TRAC transfer cask provides the structural and radiation protection of an MPC during its loading, unloading, and transfer to the storage module is approved by the USNRC in CoC No. 1032. The storage module is also designed to maintain occupational exposures ALARA during MPC transfer operations, in accordance with 10 CFR Part 20.

6.3 Shielding Design Feature

The HI-STORM UMAX System Canister Storage System shielding design is described in the FSAR, Section 1.2. The details of design are described in the licensing drawings in FSAR, Section 1.5. The HI-STORM UMAX VVM differs from above ground HI-STORM storage modules approved in CoC Nos. 72-1014 and 1032 in that the used fuel is stored below the ISFSI's top of grade (TOG). HI-STORM UMAX VVM, however, is comparable with the HI-STORM FW storage module in that it can store the certified MPC-37 and MPC-89 in the FW system. Furthermore, the HI-TRAC VW transfer cask used to transfer and install the MPCs in the storage system is identical to that used in the HI-STORM FW system. The MPC's content, conditions and the loading operations up to the time the loaded MPC's transfer cask enters at the ISFSI are identical to the HI-STORM FW system.

The HI-TRAC VW transfer cask shielding safety analyses are already provided in NRC-approved CoC No. 1032.

The HI-STORM UMAX VVM is considered an storage module and consists of a set of vertically disposed steel containers founded on a thick reinforced concrete pad and located over 20 feet below top of grade (TOP) embedded in a SES. The top region of the steel container is reinforced by a thick plate-type flange that rests on a reinforced concrete pad represented as the storage module pad.

The top opening in the container is the only location of access into the cavity and potential path of emission of radiation to the environment. The closure lid, made as an over-40 inch thick steel filled with concrete to provides maximum blockage from radiation emerging from the fuel.

The top of an MPC is equipped with a 9" thick lid and is at least about 2 feet below the bottom of the VVM Closure lid to enhance further shielding action of the lid.

Source Specification:

The applicant's design-basis source specifications for bounding calculations are identical to those used and evaluated by the staff in CoC No. 1032. The gamma source, neutron source and non-fuel hardware are the same as the sources described in HI-STORM FW FSAR.

6.3.1 Staff Evaluation

The staff performed shielding and source term calculations using SCALE 6 to compare photon and neutron sources with the applicant's evaluation and concludes that the design of the shielding system for the HI-STORM UMAX VVM system's applicable design and acceptance criteria is in compliance with 10 CFR Part 72, and that acceptance criteria have been satisfied. The staff concluded that the HI-STORM UMAX VVM system shielding system will provide reasonable assurance for safe storage of spent fuel. This finding is based on an evaluation of the FSAR and supporting documentation, compliance with the regulations and NUREG 1536, Rev. 1 guidance, along with staff confirmatory calculations and modeling analysis, and accepted engineering practices.

6.4 Shielding Analysis

The applicant used the MCNP-5 code for all of the shielding analyses. FSAR Figure 5.1.1 or Figure 1 and Figure 2 of HI-212519 i(supplied as supporting documentation) identify the locations of the dose points that are shown in FSAR Tables 5.1.1 and 5.1.2 or Tables 1 and 2 of

HI-212519 loaded with MPC-32 and MPC-37, on the surface and 1 meter from the surfaces respectively. Dose point #1 represents the side of the closure lid shell on top of the inlet plenum. The maximum dose rate is reported for the side surface of the lid shell, while the dose rate value reported at 1.0 meter was taken at the middle of the lid shell. Dose point #2 is the location of the surface of the outlet duct. Dose point #3 is positioned on the closure lid cover plate. Dose points # 4 and #5 are the locations of the outlet and inlet vents (top surface), respectively. Dose point #6 is located over a tube that would be required for the impressed current cathodic protection system (ICCPS) test station if an ICCPS is used. Dose point #7 is located over an empty VVM located adjacent to four loaded VVMs. The annual dose at 100 meters from a Single HI-STORM UMAX VVM is provided in FSAR Table 5.1.3.

The dose rate profiles across the lid and the ISFSI pad are provided in FSAR Tables 5.4.2 and 5.4.3.

6.4.1 Off-Normal Condition

The potential off-normal conditions and their effect on the HI-STORM UMAX Canister Storage System are provided in Chapter 12 of HI-STORM FW FSAR. According to the applicant, none of the off-normal conditions as defined in accordance with ANSI/ANS-57.9, listed in Chapter 12 of SAR, and as defined in Section 2.3 of SAR have any impact on the shielding analysis. Therefore for the purpose of shielding evaluation both off-normal and normal conditions are treated identically by the applicant.

6.4.2 Occupational Exposures

The applicant states that the HI-TRAC VW transfer cask provides shielding to maintain occupational exposures ALARA in accordance with 10 CFR Part 20, while also maintaining the maximum load on the plant's crane hook to below the rated capacity of the crane. The HI-TRAC VW calculated dose rates for a set of reference conditions are provided in the HI-STORM FW FSAR, Section 5.1 and evaluated by the staff in CoC No. 1032.

6.4.3 Off-Site Dose Calculation

The off-site dose for normal operating conditions to a real individual beyond the controlled area boundary is limited by 10 CFR 72.104(a) to a maximum of 25 mrem/year whole body, 75 mrem/year thyroid, and 25 mrem/year for other critical organs, including contributions from all nuclear fuel cycle operations. According to the applicant, since these limits are dependent on plant operations as well as site specific conditions, the determination and comparison of ISFSI doses to this limit are necessarily site-specific. Dose rates for a single cask on contact, at 1m, and at 100m distance using the HI-STORM UMAX Canister Storage System are provided in the FSAR, Tables 5.1.1, 5.1.2, and 5.1.3. The determination of site-specific ISFSI dose rates at the site boundary and demonstration of compliance with regulatory limits is performed by the general licensee in accordance with 10 CFR 72.212.

FSAR Table 5.1.4 presents the dose at 100 meters from a single HI-STORM UMAX storage module with MPC-32 and MPC-37 loaded with design basis fuel for accident condition which is in compliance with the 10CFR72.106.

Dose rates were also calculated at a distance of 100 m from the VVM. This would indicate what portion of the dose rate results from direct radiation through the concrete and soil of the ISFSI

pad as opposed to radiation from the streaming from the air inlet and outlet. The dose locations for the profile are shown in FSAR Figure 5.3.2, and are labeled alphabetically (A through X). The calculated dose rates are listed in FSAR Tables 5.4.2 and 5.4.3 for MPC-32 and MPC-37, respectively.

6.4.4 Staff Evaluation

The staff performed confirmatory analyses of selected dose rates using the MAVRIC sequence of the SCALE 6 code system, with the Monaco three dimensional Monte Carlo shielding analysis code. The staff based its evaluation on the design features and model specifications presented in the drawings shown in SAR. Limiting fuel characteristics, and the burnup and cooling time, are included in the TS, as are the dose rates profile across the HI-STORM UMAX lid and surrounding ISFSI pad. The staff's calculated dose rates were in reasonable agreement with the SAR values or were generally lower due to the applicant's conservative loading assumptions. The staff found that the SAR has adequately demonstrated that the HI-STORM UMAX is designed to meet the criteria of 10 CFR 72.104(a) and 72.106.

Each general licensee is responsible to verify compliance with 10 CFR 72.104(a) in accordance with 10 CFR 72.212. In addition, a general licensee will also have an established radiation protection program as required by 10 CFR Part 20, Subpart B and will demonstrate compliance with dose limits to individual members of the public and workers including for excavation activities, as required, by evaluation and measurements. The staff notes that the system contents result in relatively significant direct radiation dose rates, which is a concern primarily for operations involving the transfer cask such as, loading, unloading, and transport for the UMAX system. Thus, each user may be required to take additional ALARA precautions to minimize doses to personnel and to make additional use of realistic fuel characteristics and distances to demonstrate compliance with public dose limits in 10 CFR Part 20 and 10 CFR Part 72. As explained above, the staff reviewed the accident evaluation and found it acceptable for the application. The staff has reasonable assurance that the direct radiation from the UMAX satisfies 10 CFR 72.106(b) at or beyond a controlled boundary of 100 meters from the design-basis accidents.

6.5 Evaluation Findings

Based on the NRC staff's review of information provided for the HI-STORM UMAX Canister: Storage System application, the staff finds the following:

F6.1 The FSAR sufficiently describes shielding design features and design criteria for the structures, systems, and components important to safety.

F6.2 Radiation shielding features of the HI-STORM UMAX Canister Storage System are sufficient to meet the radiation protection requirements of 10 CFR Part 20, 10 CFR 72.104, and 10 CFR 72.106.

F6.3 Operational restrictions to meet dose and ALARA requirements in 10 CFR Part 20, 10 CFR 72.104 and 72.106 are the responsibility of each general licensee. The HI-STORM UMAX Canister Storage System shielding features are designed to satisfy these requirements.

F6.4 The staff finds the design addresses construction activities involving excavation (for ISFSI expansion) adjacent to the (operating) HI STORM UMAX Canister Storage System sufficient to ensure that the shielding features will continue to be sufficient to meet the radiation protection requirements of 10 CFR Part 20, 10 CFR 72.104, and 10 CFR 72.106.

The staff concludes that the design of the radiation protection system of the HI-STORM UMAX Canister Storage System can be operated in compliance with 10 CFR Part 72 and that the applicable design and acceptance criteria have been satisfied. The evaluation of the radiation protection system design provides reasonable assurance that the HI-STORM UMAX Canister Storage System will provide safe storage of spent fuel. This finding is based on a review that considered the regulation itself, the appropriate regulatory guides, applicable codes and standards, the applicant's analyses, the staff's confirmatory analyses, and acceptable engineering practices.

7 CRITICALITY EVALUATION

The criticality review and evaluation ensures that SNF to be placed into the DSS remains subcritical under normal, off-normal, and accident conditions involving handling, packaging, transfer, and storage. The criticality review is designed to fulfill the strategic outcome of no inadvertent criticality events, part of the strategic goal of safety described in the NRC's strategic plan (NUREG-1614).

For the staff's criticality evaluation, the staff determined that the HI-STORM UMAX Canister Storage System design is identical to the NRC approved HI-STORM FW System in CoC No. 1032. Criticality safety was demonstrated during the CoC No. 1032 review, and there are no substantive differences applicable to the criticality evaluation between the two systems except for the storage module configuration, which is now below ground.

As part of the review, the staff asked the applicant whether flooding of the MPC is a plausible scenario. The applicant identified two unlikely independent events that would need to occur before a criticality event could be possible - a flooding of the CEC concurrent with a failure of the MPC confinement boundary. Under normal conditions, the interior of the MPC is dry and the inside of the CEC is dry. Since during storage conditions the HI-STORM UMAX Canister Storage System is dry within the MPC, the maximum k_{eff} is significantly below the limiting maximum k_{eff} of 0.95 as demonstrated in the CoC No. 1032 evaluation.

The applicant identified several scenarios that could lead to accumulation of water in the CEC, including underground sources of water and precipitation, and described the defense-in-depth of each scenario in their response to request for information (RAI) 6-1. The applicant also addressed the potential for water to leak into the MPC assuming the failure of the CEC to remain dry in this RAI response, and indicated that the welding procedure to ensure the inert atmosphere (i.e., helium) remains in the MPC; the materials, manufacturing processes, closure procedure, and long term integrity of the MPC under storage conditions; and the physical protection of the MPC due to the underground configuration of the HI-STORM UMAX Canister Storage System, combined to make the potential for the MPC to allow an ingress of water to be highly unlikely. The staff finds that failure of the MPC as identified by the applicant is highly unlikely. Staff is, however, continuing to evaluate the generic issue of flooding through a generic issue process.

The staff reviewed the applicant's criticality safety analyses and determined that the applicant's assessment on system criticality safety is consistent with the new review guidance of NUREG-1536. Based on its review, the staff finds the criticality safety assessment acceptable and that there is reasonable assurance that the HI-STORM UMAX Canister Storage System meets the regulatory requirements of 10 CFR Part 72 and the acceptance criteria specified for both intact and damaged fuel.

7.1 Evaluation Findings

In summary, the staff finds the following:

F7.1 Structures, systems, and components ITS are described in sufficient detail in Chapters 1, 2 and 6 of the HI-STORM UMAX Canister Storage System and HI-STORM FW MPC Storage System FSARs to enable an evaluation of their effectiveness.

F7.2 The staff determined the cask and its spent fuel transfer systems are designed to be subcritical under all credible conditions.

F7.3 The staff determined the criticality design is based on favorable geometry, and fixed neutron poisons. An appraisal of the fixed neutron poisons has shown that they will remain effective for the term requested in the application. The staff determined there is no credible way for the fixed neutron poisons to significantly degrade during the requested term in the application; therefore, there is no need to provide a positive means to verify their continued efficacy as required by 10 CFR 72.124(b).

F7.4 The staff determined the applicant's analysis and evaluation of the criticality design and performance have demonstrated that the cask will enable the safe storage of spent fuel with respect to criticality safety for the term requested in the application.

The staff concludes that the criticality design features for the HI-STORM UMAX Canister Storage System are in compliance with 10 CFR Part 72, and that the applicable design and acceptance criteria have been satisfied. The evaluation of the criticality design provides reasonable assurance that the HI-STORM UMAX Canister Storage System will allow safe storage of spent fuel. These findings are reached on the basis of a review that considered the regulation itself, NUREG 1536, Rev.1, applicable codes and standards, and accepted engineering practices.

8 MATERIALS EVALUATION

The materials review ensures adequate material performance of components ITS of the HI-STORM UMAX Canister Storage System, including the spent fuel canister or cask, under normal, off-normal, and accident-level conditions. To ensure an adequate margin of safety in the design basis of the HI-STORM UMAX Canister Storage System, the review determined that there is reasonable assurance that:

- The physical, chemical, and mechanical properties of materials for components ITS meet their service requirements including normal, off normal and accident-level conditions.

- Materials for components ITS have sufficient requirements to control the quality of the production, fabrication, and test activities.

- Materials for ITS components are selected to accommodate the effects of, and to be compatible with, the ISFSI site characteristics, environmental conditions, and duration of the license period.

- The SNF cladding is protected from gross rupture and from conditions that could lead to fuel redistribution.

- The HI-STORM UMAX Canister Storage System is designed to maintain the spent fuel in a readily retrievable condition.

- Other materials which support or protect ITS components (such as coatings) are suitable for the application.

The staff states the HI-STORM UMAX Canister Storage System incorporates the use of Metamic HT for the fuel basket structure. The remaining materials used in the fabrication of the HI-STORM UMAX Canister Storage System have been used in previously staff-reviewed storage system designs. There are no changes to the various other materials used to fabricate the HI-STORM UMAX Canister Storage System from those previously reviewed and approved in other storage system designs, therefore evaluation is provided for the HI-STORM UMAX VVM. The HI-STORM UMAX Canister Storage System VVM is similar to the HI-STORM 100U VVM approved in CoC No. 72-1014. The major differences between the HI-STORM UMAX and HI-STORM 100U VVM are that the HI-STORM UMAX VVM cavity is larger in diameter and the HI-STORM UMAX closure lid features a modified outlet ventilation duct system. In addition, the HI-STORM UMAX VVM is the underground equivalent of the HI-STORM FW overpack approved in CoC No. 72-1032.

8.1 HI-STORM UMAX Canister Storage System Materials

8.1.1 Metamic HT Spent Fuel Basket:

Metamic HT is a Holtec proprietary aluminum-based material intended for dual purpose use in the HI-STORM UMAX spent fuel basket. Metamic HT is designed to be both a neutron poison for criticality control and a load-bearing structural material. Metamic-HT is a powder metallurgy material composed of aluminum combined with aluminum oxide and boron carbide (Holtec uses the terminology metal matrix composite (MMC) to generically describe Metamic HT). The aluminum oxide is a finely dispersed second-phase particle that provides enhanced room temperature and elevated temperature (creep) strength. The boron carbide is the neutron poison used for criticality control. The neutronic properties of Metamic HT are essentially identical to previously reviewed classical Metamic. The staff finds the composition and properties of Metamic HT to be unique; however the applicants test program was comprehensive in scope and supported the wide variety of property data for characterizing Metamic HT. Using the guidance of ASME Code, Section II, Appendix 5, Holtec determined mechanical properties at various temperatures. Material properties are discussed in Section 8.4 of the application. The staff reviewed all materials selected and determined that they are acceptable and provide reasonable assurance for safety of the package based on specifications, temperature dependent mechanical properties, including yield strength, tensile

strength, allowable strength, modulus of elasticity, and coefficient of thermal expansion conforming to ASME Code requirements.

METAMIC-HT CORROSION RESISTANCE:

The applicant concluded that Metamic HT has more than adequate corrosion resistance for the intended service. Metamic HT was tested for compatibility with borated water, as would be typical for cask loading and unloading conditions. Aluminum alloys are very slightly corroded by borated water and Metamic HT performed similar to other aluminum-based materials in immersion tests. Based upon the applicant's anslysis, the staff confirms that Metamic HT is not susceptible to chemical and/or galvanic reactions under conditions discussed above, , its dry, inert environment during storage, and its isolated contact with surrounding internal canister materials.

Thus, the staff concludes that applicant's use of Metamic HT is in accordance with 10 CFR 72.120(d), and that the applicant has met the requirements of 10 CFR 72.124 in that the . materials used for criticality control are adequately designed and specified to perform their intended function.

8.1.2 Cavity Enclosure Container (CEC) Portion of the Vertical Ventilation Module (VVM):

The CEC is discussed in Section 1.2.2 and corrosion mitigation in Section 8.7 of the application. The staff states the CEC portion of the VVM is not part of the MPC containment boundary, and may be exposed to the extent where localized in-leakage of groundwater occurs.

The applicant evaluated soil corrosivity, a "10 point" soil-test evaluation procedure, in accordance with ANSI, Appendix A, standard for Polyethylene Encasement for Ductile-Iron Pipe Systems", ANSI/American Water Works Association (AWWA) C105/A21. The soil evaluation criteria in this standard focuses on parameters such as: resistivity, pH, redox (oxidation-reduction) potential, sulfides, moisture content, potential stray current, and experience with existing installations in the area. ISFSI soil environment corrosivity is categorized as either "mild" for a soil test evaluation resulting in 9 points or less or "aggressive" for a soil test evaluation resulting in 10 points or greater. For mild corrosivity an exterior coating with either concrete encasement or cathodic protection or both is used and for aggressive corrosivity an exterior coating with cathodic protection is used, concrete encasement is optional.

Based upon the information presented in the application, the staff concludes that the VVM is an ITS buried structure and susceptible to distinctive corrosion conditions as compared to an above-ground steel structure. Corrosion mitigation of the exterior of the CEC necessitates consideration due to inaccessibility of the exterior coated surface following installation, potential for a highly corrosive soil environment at certain sites, and potential for a high radiation field. The buried configuration will not allow for the inspection and re-application of surface preservation. The staff finds, based on the 10 point test discussed above, Appendix A of the FSAR, which provides procedures for soil survey tests, observations and their interpretation, provides reasonable assurance as part of corrosion mitigation measures that the VVM will meet intended design life and perform intended safety functions.

8.1.3 Coatings:

Coatings are discussed in Section 8.7 of the FSAR. The applicant states that in addition to a corrosion allowance for the CEC structural steel itself, the CEC is coated with a radiation resistant surface preservative designed for below-grade and/or immersion service. Inorganic and/or metallic coatings are sufficiently radiation resistant for this application; therefore radiation testing is not required for inorganic or metallic coatings. Organic coatings such as epoxy, however, must have proven radiation resistance or must be tested without failure to at least 1x10E7 Rad. Radiation resistance to lower radiation levels is acceptable on a site-specific basis.

The applicant states that radiation testing is performed in accordance with ASTM D 4082, "Standard Test Method for Effects of Gamma Radiation on Coatings for Use in Light Water Nuclear Power Plants", or equivalent. The coating should be conservatively treated as a service Level II coating as described in USNRC Regulatory Guide 1.54. As such, the coating will be subjected to appropriate quality assurance in accordance with the applicable guidance provided by ASTM D 3843-00, "Standard Practice for Quality Assurance for Protective Coatings Applied to Nuclear Facilities".

The applicant states that the preference is for the coating to be shop applied in accordance with manufacturer instructions and, if appropriate, applicable guidance from ANSI C 210-03, "Standard Practice for Liquid-Epoxy Coating Systems for the Interior and Exterior of Steel Water Pipelines" will be used. A Keeler & Long polyamide-epoxy coating, according to the manufacturer's product data sheet, is pre-tested to radiation levels up to 1x10E9 Rad without failure.

Alternative coatings may be selected by Holtec on the basis of pre-established criteria which are provided in FSAR Chapter 8. These criteria include consideration of various environmental conditions along with a ranking of their relative importance. The Keeler & Long epoxy meets all the criteria and is the standard coating for use.

8.1.4 Concrete Encasement

Concrete encasement is discussed in Section 8.7 of the FSAR. The applicant states that the CEC concrete encasement will provide a minimum of 5 inches of cover to provide a pH buffering effect for additional corrosion mitigation. The 5-inch minimum thickness is conservative when compared to ACI 318, which recommends up to 3 inches of concrete cover over steel reinforcement in aggressive environments. The concrete encasement is restricted to mild soil environments (unless used in conjunction with cathodic protection) and has a non-structural role. The applicant asserts that the 5-inch concrete encasement thickness is considered more than sufficient to provide reasonable assurance that a 60-year service life can be achieved. Regardless of reinforcement method, according to the application, the material selected will be corrosion-resistant or otherwise appropriately coated (e.g., epoxy coated steel wire) for corrosion resistance. The concrete encasement shall be installed in accordance with guidance from ACI 318 for commercial concrete). Installation procedures will address mix designs, testing, mixing, placement, and reinforcement, with the aim to enhance concrete durability and minimize voids and micro-cracks.

The staff states that this design, as described above, provides for conservative assumptions which the staff finds acceptable because the thickness specified for the concrete in TS Appendix B, Table 3-4, is greater than that specified by several recognized codes or references

that are based upon a 20 year minimum design life. Thus, a working life of greater than 20 years is reasonably assured.

Reinforcement is often avoided in structures where the primary purpose is radiation shielding because the presence of reinforcement bars can create unintended voids in the concrete, leading to a deficient radiation shield. However, in the case of the CEC, the primary shielding is accomplished by the soil backfill. The staff finds that the purpose of the HI-STORM UMAX concrete encasement is to mitigate any corrosive effects from the soil, not provide for radiation shielding.

8.1.5 Impressed Current Cathodic Protection System (ICCPS)

ICCPS is discussed in Section 8.7 of the application. According to the applicant, if the subgrade around the CEC is highly aggressive and warrants an ICCPS then the user may choose to either extend an existing ICCPS to protect the installed ISFSI, or to establish an autonomous system. The application indicates that he initial startup of the ICCPS is to occur within one year after installation of the VVM to ensure timely corrosion mitigation. In addition, the application indicates that the ICCPS should be maintained operable at all times after initial startup except for system shutdowns due to power outages, repair or preventive maintenance and testing, or system modifications.

Although there are a multitude of ISFSI variables that will affect the design of the ICCPS for a particular site, the essential criteria for its performance and operational characteristics are established in the FSAR and will be applied by users as required by 10 CFR 72.212(b)(2)(ii)(3).

Based upon the information described above, the staff finds that the ICCPS provides reasonable assurance that the aggressive corrosion conditions of some soils will not cause degradation of the CEC (including the bottom plate) to the extent that the CEC structural integrity is affected or allow in-leakage of ground water into the storage cavity. The staff finds that the surveillance and maintenance programs outlined in the FSAR are acceptable for establishing the continued integrity of the CEC because they are consistent with industry and ASME Code guidelines. Additionally, the staff's conclusion is based upon appropriate TS requirements in Appendix B, Section 3 that have been established for the ICCPS.

8.1.6 Other Materials of Construction

The balance of the HI-STORM UMAX Canister Storage System is fabricated from materials which have all been previously evaluated by the staff for their suitability in storage applications. The bill of materials in FSAR drawings and FSAR Chapter 8 provide details of each material type and specification. All the materials have been previously reviewed and employed for staff approved 10 CFR Part 71 transportation CoCs, but a brief discussion of materials related findings from prior staff evaluations are summarized below for information.

8.1.7 Confinement Boundary

The fuel canister confinement is fabricated from one of several ASME grades of austenitic stainless steel, referred to by the applicant as "Alloy X". Alloy X assumes the least favorable property characteristics from among the several materials grades specified. These properties are used for all design calculations. The purpose of assuming the least favorable property

characteristics from among the several material grades specified is to allow for free interchange of the several grades of stainless steel. This provides the applicant with procurement flexibility while complying with all required design properties. This method of allowing for material substitution has been previously reviewed by the staff in CoC's Nos. 1014 and 1032 and found to be acceptable. The use of austenitic stainless steel also means that the fuel canister is resistant to brittle fracture issues.

8.1.8 Gamma and Neutron Shield

As described by the applicant, steel, concrete, and the subgrade are the principal shielding materials in the HI-STORM UMAX. The steel and concrete shielding materials in the Closure lid provide additional gamma and neutron attenuation to reduce dose rates. Steel, lead, and water are the principal shielding materials in the HI-TRAC transfer cask. The combination of these three shielding materials ensures that the radiation and exposure objectives of 10CFR72.106 and ALARA are met. As explained by the applicant, the extent and location of shielding in the transfer cask plays an important role in minimizing the personnel doses during loading, handling, and transfer. The MPC fuel basket structure provides the initial attenuation of gamma and neutron radiation emitted by the radioactive contents. The MPC shell, baseplate, and thick lid provide additional gamma attenuation to reduce direct radiation.

Gamma and neutron shield materials in the HI-STORM UMAX VVM system are discussed in Section 1.2. The primary shielding materials used in the HI-STORM UMAX VVM system, as listed in Table 8.1.3, are plain concrete, reinforced concrete, and steel. The plain concrete provides the main shielding function in the HI-STORM UMAX lids to minimize sky shine.

Shielding materials are discussed in Section 1.2 of the application. The applicant states that the radiation shield (concrete over pack) is composed primarily of un-reinforced concrete with a carbon steel liner plate on the inside. Since the overpack has no structural role, the applicant asserts that the lack of reinforcing steel is not a detriment. The lack of reinforcing steel is a deliberate exclusion in order to avoid the possibility of interior voids in the concrete which would degrade the shielding performance. This type of construction has been approved by the staff previously in CoC No. 1014 and found to be satisfactory in service at numerous installations. For the design service conditions, the applicant stated there are no conditions that will result in a degradation of the materials performance for the duration of the license period. Based upon prior staff approvals and industry experience with these materials, the staff finds that there is reasonable assurance that the materials will perform their intended function for 40 years of service with no loss of performance or adverse degradation for these materials and this specific design.

8.1.9 Weld Material

The applicant has stated that weld filler materials utilized in the welding of the confinement boundary comply with the provisions of the appropriate ASME subsection. Non-code welds (e.g. not important to safety) are made using weld procedures that meet the ASME Code Section IX, AWS D1.1, D.1.2 or equivalent. Non-destructive examinations comply with ASME Code, Section V, with acceptance criteria as specified by the code of construction for the specific component. The staff finds this acceptable as it meets the guidance of NUREG 1536, Rev. 1, and is consistent with practices previously approved by the staff in CoC No. 1032.

8.1.10 Chemical, Galvanic, or Other Reactions

The applicant has stated that the MPC is dried and helium backfilled to eliminate any credible corrosion from moisture and oxidizing gasses. Therefore, chemical, galvanic or other reactions involving the cask materials are minimal or unlikely. Because the HI-STORM UMAX MPCs employ materials that are compatible with wet and dry SNF loading and unloading operations and facilities, the staff finds that the applicant has met the requirements of 10CFR 72.236(h). The staff confirmed the designed features and concluded that these materials will not degrade over time or react with one another during environmental conditions of storage.

8.2 CORROSION MITIGATION

According to the application, the corrosion mitigation methods described in the FSAR have a support role to an ITS system (the CEC portion of the VVM) and are required as a result of the unique design features and corrosion environment associated with underground structures. Since the ITS portions of the CEC are not normally accessible for routine inspection, certain parameters of the cathodic protection system are incorporated into the technical specifications. This ensures, through operational monitoring, that the cathodic protection system is performing as designed and that no degradation of the CEC is occurring. The application explains that operational history becomes the alternative to direct inspection, hence the requirement for technical specification control of an otherwise not-safety-related system. In the event of unforeseen questions about the efficacy of the cathodic protection system (or other component of the corrosion mitigation measures) the CEC structure may be examined by means of ultrasonic inspection (UT) from the inside of a CEC cell where there is no fuel canister yet installed, by remote means in a cell where a spent fuel canister is installed, or, a cell from which the canister has been removed to allow inspection.

The staff finds that the applicant has specified in sufficient detail the design and operational parameters for an effective corrosion mitigation program for a range of potential environments. Additionally, operation and control of a cathodic protection system is placed into TS Appendix B, Section 3, to ensure reliable operation of this system in the place of a routine inspection of the protected ITS components of the CEC.

8.3 Conclusion: (Other Materials of Construction)

Since the materials of construction for the balance of the storage canister have been previously reviewed for acceptability, and since the conditions of use are unchanged, the staff finds that the materials are acceptable for their specified uses.

The HI-STORM UMAX Canister Storage System and HI-STORM FW Cask System FSARs, Chapter 8, adequately describes the materials used for SSCs ITS and the suitability of those materials for their intended functions is sufficient detail to evaluate their effectiveness.

The applicant has met the requirements of 10 CFR 72.122(a). The material properties of SSCs important to safety conform to quality standards commensurate with their safety function.

The applicant has met the requirements of 10 CFR 72.122(h)(1) and 72.236(h). The design of the DSS and the selection of materials adequately protect the SNF cladding against degradation that might otherwise lead to damaged fuel.

The applicant has met the requirements of 10 CFR 72.236(h) and 72.236(m). The material properties of SSCs ITS will be maintained during normal, off-normal, and accident conditions of operation so the SNF can be readily retrieved without posing operational safety problems.

The applicant has met the requirements of 10 CFR 72.236(g). The material properties of SSCs ITS will be maintained during all conditions of operation so the SNF can be safely stored for the minimum required years and maintenance can be conducted as required.

8.4 Evaluation Findings

F8.1 The staff concludes the material properties of the structures, systems, and components of the HI-STORM UMAX Canister Storage System are in compliance with 10 CFR Part 72, and that the applicable design and acceptance criteria have been satisfied. The evaluation of the material properties provides reasonable assurance the cask will allow safe storage of SNF. This finding is reached on the basis of a review that considered the regulation itself, appropriate regulatory guides, applicable code requirements and standards, and accepted engineering practices.

9 OPERATING PROCEDURES EVALUATION

The operating procedures review ensures that the applicant's FSAR presents acceptable operating sequences, guidance, and generic procedures for identified key operations. The review also ensures that the FSARs incorporate and are compatible with the applicable operating control limits in the TS.

9.1 Areas of Review

The HI-STORM UMAX and HI-STORM FW FSARs were reviewed and the following operations were acceptably addressed:

Loading Operations

- Fuel Specifications
- Damaged Fuel
- Subcriticality Features
- ALARA
- Offsite Releases
- Draining and Drying
- Filling and Pressurization
- Welding and Sealing
- Administrative Programs

Cask Handling and Storage Operations

Cask Unloading

- Damaged Fuel
- Cooling, Venting, and Reflooding
- Fuel Crud
- ALARA
- Offsite Release

9.2 Staff Evaluation

The staff concludes that the generic procedures and guidance for operation of the HI-STORM UMAX Canister Storage System are in compliance with 10 CFR Part 72 and that the applicable acceptance criteria have been satisfied. The evaluation of the operating procedure descriptions provided in the FSARs offer reasonable assurance that the system will enable safe storage of spent fuel. This finding is based on a review that considered the regulations, appropriate regulatory guides, applicable codes and standards, and acceptable practices.

9.3 Evaluation Findings

F9.1 The HI-STORM UMAX Canister Storage System can be wet loaded and unloaded. General procedure descriptions for these operations are summarized in Chapter 9 of the HI-STORM UMAX Canister Storage System and HI-STORM FW Cask System FSARs. Detailed procedures are developed and evaluated by general licensees on a site-specific basis.

F9.2 The welded MPC allows for ready retrieval of the spent fuel for further processing or disposal as required.

F9.3 The general operating procedures are designed to prevent contamination of the MPCs and facilitate decontamination of the storage module. Routine decontamination will be necessary after the cask is removed from the spent fuel pool.

F9.4 No significant radioactive effluents are produced during storage. Any radioactive effluents generated during the cask loading and unloading will be governed by the 10 CFR Part 50 license conditions.

F9.5 The general operating procedures described in the FSARs are adequate to protect health and minimize danger to life and property. Detailed procedures are developed and evaluated by general licensees at their sites.

F9.6 Section 11 of the FSAR assesses the operational restrictions to meet the limits of 10 CFR Part 20. Additional site-specific restrictions may also be established by the site licensee.

F9.7 The staff concludes that the generic procedures and guidance for the operation of the UMAX Canister Storage System are in compliance with 10 CFR Part 72 and that the applicable acceptance criteria have been satisfied. The evaluation of the operating procedure descriptions provided in the FSARs offers reasonable assurance that the cask will enable safe storage of spent fuel. This finding is based on a review that considered the regulations, appropriate regulatory guides, applicable codes and standards, and accepted practices.

10 ACCEPTANCE TESTS AND MAINTENANCE PROGRAM

The specific discipline acceptance tests and maintenance programs are evaluated by specific disciplines in sections 3 through 8 of this SER. The results of the evaluation are captured in their applicable sections.

11 ACCIDENT ANALYSIS EVALUATION

The specific discipline accident analyses are evaluated in sections 3 through 9 of this SER. The results of the evaluation are captured in their applicable sections.

12 TECHNICAL SPECIFICATION S AND OPERATING CONTROLS AND LIMITS EVALUATION

12.1 Objective

The technical specifications and operating controls and limits review ensures that the operating controls and limits of the TS, including their bases and justification, meet the requirements of 10 CFR Part 72. The evaluation is based on information provided by the applicant in the HI-STORM UMAX Canister Storage System and HI-STORM FW MPC System FSARs Chapter 13 as well as accepted practices and any commitments discussed in other chapters of the FSARs.

12.2 Evaluation Findings

F.12.1 The staff concludes that the conditions for use of the HI-STORM Canister Storage System identify necessary TS to satisfy 10 CFR Part 72 and that the applicable acceptance criteria have been satisfied. The TS provide reasonable assurance that the cask will provide for safe storage of spent fuel. This finding is reached on the basis of a review that considered the regulation itself, appropriate regulatory guides, applicable codes and standards, and accepted practices.

13 QUALITY ASSURANCE EVALUATION

The purpose of this review and evaluation is to determine whether Holtec has a quality assurance (QA) program that complies with the requirements of 10 CFR Part 72, Subpart G. Holtec's QA program has been previously evaluated in the review of the HI-STORM 100 Cask system, CoC No. 1014, and HI-STORM FW MPC System, CoC No. 1032, applications and subsequent amendments.

13.1 Areas Reviewed

QA Organization
QA Program
Design Control
Procurement Document Control
Instructions, Procedures, and Drawings
Document Control
Control of Purchased Material, Equipment, and Services
Identification and Control of Materials, Parts, and Components
Control of Special Processes
Licensee Inspection
Test Control
Control of Measuring and Test Equipment
Handling, Storage, and Shipping Control
Inspection, Test, and Operating Status

Nonconforming Materials, Parts, or Components
Corrective Action
QA Records
Audits
NUREG-1536, Revision 1 provides the criteria for evaluating the above 18 areas. In a number of cases, the description of, or actions to be taken by, personnel involved in quality activities were incorporated by reference to the applicable sections of the Holtec's Quality Assurance Manual (HQAM). It was therefore necessary to review such referenced sections in the HQAM to determine whether the QA program, as submitted, met the requirements of 10 CFR Part 72, Subpart G. While this evaluation determined that the QA program is acceptable, proper implementation of the QA program will be assessed during future NRC inspections.

13.2 Evaluation Findings

F13.1 The QA program describes the requirements, procedures, and controls that, when properly implemented, comply with the requirements of 10 CFR Part 72, Subpart G, and 10 CFR Part 21, Reporting of Defects and Noncompliance.

F13.2 The structure of the organization and assignment of responsibility for each activity ensure that designated parties will perform the work to achieve and maintain specified quality requirements.

F13.3 Conformance to established requirements will be verified by qualified personnel and groups not directly responsible for the activity being performed. These personnel and groups report through a management hierarchy which grants the necessary authority and organizational freedom and provides sufficient independence from economic and scheduling influences.

14 CONCLUSIONS

Based on its review of the FSAR and supporting documentation, the staff has determined that there is reasonable assurance that: (i) the activities authorized by the HI-STORM Canister Storage System, CoC No. 1040 can be conducted without endangering the health and safety of the public and (ii) these activities will be conducted in compliance with the applicable regulations of 10 CFR Part 72. The staff has further determined that the issuance of the amendment will not be inimical to the common defense and security. Therefore, CoC No. 1040 should be approved.

Principle contributors: Jeremy Smith, Dr. Jorge Solis, Dr. Jimmy Chang, Eli Goldfiez, Jason Piotter, David Tarantino, John Goshen, P.E.

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12073

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NUCLEAR REGULATORY COMMISSION

10 CFR Part 72

[NRC–2014–0120]

RIN 3150–AJ42

List of Approved Spent Fuel Storage Casks: Holtec International HI–STORM Underground Maximum Capacity Canister Storage System, Certificate of Compliance No. 1040

AGENCY: Nuclear Regulatory Commission.

ACTION: Final rule.

SUMMARY: The U.S. Nuclear Regulatory Commission (NRC) is amending its spent fuel storage regulations by adding the Holtec International HI–STORM Underground Maximum Capacity (UMAX) Canister Storage System, Certificate of Compliance (CoC) No. 1040, to the ''List of approved spent fuel storage casks.'' Holtec International's intent with this design is to provide an underground storage option compatible with the Holtec International HI–STORM FLOOD/WIND (FW) System (CoC No. 1032).

DATES: This final rule is effective on April 6, 2015.

ADDRESSES: Please refer to Docket ID NRC–2014–0120 when contacting the NRC about the availability of information for this action. You may obtain publicly-available information related to this action by any of the following methods:

- *Federal Rulemaking Web site:* Go to *http://www.regulations.gov* and search for Docket ID NRC–2014–0120. Address questions about NRC dockets to Carol Gallagher; telephone: 301–415–3463; email: *Carol.Gallagher@nrc.gov.* For technical questions, contact the individual listed in the **FOR FURTHER INFORMATION CONTACT** section of this document.
- *NRC's Agencywide Documents Access and Management System (ADAMS):* You may obtain publicly-available documents online in the ADAMS Public Documents collection at *http://www.nrc.gov/reading-rm/adams.html.* To begin the search, select ''ADAMS Public Documents'' and then select ''*Begin Web-based ADAMS Search.''* For problems with ADAMS, please contact the NRC's Public Document Room (PDR) reference staff at 1–800–397–4209, 301–415–4737, or by email to *pdr.resource@nrc.gov.* For the convenience of the reader, instructions about obtaining materials referenced in this document are provided in the ''Availability of Documents'' section.
- *NRC's PDR:* You may examine and purchase copies of public documents at the NRC's PDR, Room O–1F21, One White Flint North, 11555 Rockville Pike, Rockville, Maryland 20852.

FOR FURTHER INFORMATION CONTACT: Gregory R. Trussell, Office of Nuclear Material Safety and Safeguards, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, telephone: 301–415–6445, email: *Gregory.Trussell@nrc.gov.*

SUPPLEMENTARY INFORMATION:

Table of Contents

I. Background

Section 218(a) of the Nuclear Waste Policy Act (NWPA) of 1982, as amended, requires that ''the Secretary [of the Department of Energy] shall establish a demonstration program, in cooperation with the private sector, for the dry storage of spent nuclear fuel at civilian nuclear power reactor sites, with the objective of establishing one or more technologies that the [Nuclear Regulatory] Commission may, by rule, approve for use at the sites of civilian nuclear power reactors without, to the maximum extent practicable, the need for additional site-specific approvals by the Commission.'' Section 133 of the NWPA states, in part, that ''[the Commission] shall, by rule, establish procedures for the licensing of any technology approved by the Commission under Section 219(a) [sic: 218(a)] for use at the site of any civilian nuclear power reactor.''

To implement this mandate, the Commission approved dry storage of spent nuclear fuel in NRC-approved casks under a general license by publishing a final rule in part 72 of Title 10 of the *Code of Federal Regulations* (10 CFR), which added a new subpart K within 10 CFR part 72 entitled, ''General License for Storage of Spent Fuel at Power Reactor Sites'' (55 FR 29181; July 18, 1990). This rule also established a new subpart L within 10 CFR part 72 entitled, ''Approval of Spent Fuel Storage Casks,'' which contains procedures and criteria for obtaining NRC approval of spent fuel storage cask designs.

The NRC published a direct final rule on this amendment in the **Federal Register** on September 9, 2014 (79 FR 53281). The NRC also concurrently published an identical proposed rule on September 9, 2014 (79 FR 53352). The NRC received at least one comment that is treated as a significant adverse comment on the proposed rule; therefore, the NRC withdrew the direct final rule on November 19, 2014 (79 FR 68763), and is proceeding, in this document, to address the comments on the proposed rule (see Section III, Public Comment Analysis, of this document).

II. Discussion of Changes

By letter dated June 29, 2012, and as supplemented on July 16 and November 20, 2012; January 30, April 2, April 19, June 21, August 28, December 6, and December 31, 2013; and January 13, and January 28, 2014, Holtec International submitted an application to add the HI–STORM UMAX Canister Storage System to the list of approved spent fuel storage casks in 10 CFR part 72. The HI–STORM UMAX Canister Storage System is a spent fuel storage system designed to be in full compliance with the requirements of 10 CFR part 72. Holtec International's intent with this design is to provide an underground storage option compatible with the Holtec International HI–STORM FW System as described in the Final Safety Analysis Report (FSAR) for the HI–STORM FW

System. The underground structure system is described in the FSAR for the HI–STORM UMAX Canister Storage System. The HI–STORM UMAX Canister Storage System stores a hermetically sealed canister containing spent nuclear fuel (SNF) in an in-ground vertical ventilated module (VVM). The HI–STORM UMAX Canister Storage System is designed to provide long-term underground storage of loaded multi-purpose canisters (MPC) previously certified for storage in CoC No. 1032. The HI–STORM UMAX VVM is the underground equivalent of the HI–STORM FW storage module. Although the storage cavity dimensions and the air ventilation system in the HI–STORM UMAX VVM have been selected to enable it to also store all MPCs certified for storage in the HI–STORM 100 storage module, CoC No. 1040 does not approve the storage of all MPCs certified for storage in the HI–STORM 100 storage module in the HI–STORM UMAX VVM at this time. The HI–STORM UMAX Canister Storage System can store either Pressurized Water Reactor or Boiling Water Reactor fuel assemblies in the MPC–37 or MPC–89 models, respectively. The number associated with the MPC is the maximum number of fuel assemblies the MPC can contain in the fuel basket. The external diameters of the MPC–37 and MPC–89 are identical to allow the use of a single storage module design, however the height of the MPC, as well as the storage module and transfer cask, are variable based on the SNF to be loaded.

As documented in the safety evaluation report (SER), the NRC staff performed a detailed safety evaluation of the proposed CoC request submitted by Holtec International.

The HI–STORM UMAX Canister Storage System, when used under the conditions specified in the CoC, the Technical Specifications (TSs), and the NRC's regulations, will meet the requirements of 10 CFR part 72; therefore, adequate protection of public health and safety will continue to be ensured. When this final rule becomes effective, persons who hold a general license under 10 CFR 72.210 may load spent nuclear fuel into HI–STORM UMAX Canister Storage Systems that meet the criteria of CoC No. 1040 under 10 CFR 72.212.

III. Public Comment Analysis

The NRC received multiple comments from private citizens on the companion proposed rule to the direct final rule published on September 9, 2014. The NRC has not made any changes to the proposed rule as a result of the public comments the NRC has received.

Summary of Comments

The NRC received almost a dozen comments on the proposed rule, many raising multiple and overlapping issues. Because the NRC received at least one comment that it is treating as a significant adverse comment on the proposed rule (raising issues the NRC deemed serious enough to warrant a substantive response to clarify the record), the NRC withdrew the direct final rule and is responding to the comments here. Other comments were not treated as significant adverse comments because, in most instances, they were beyond the scope of this rulemaking. Nonetheless, in addition to responding to the issues raised in the comments treated as significant adverse comments, the NRC is also taking this opportunity to respond to some of the issues raised in the comments that are beyond this scope of this rulemaking in order to clarify information about the CoC rulemaking process related to the comments received.

Aging Management Programs

Many of the comments the NRC received questioned the fact that aging management programs (AMPs) were not being established for this CoC system. Commenters noted that the NRC has not yet issued the revision to NUREG–1927 (''Standard Review Plan for Renewal of Spent Fuel Dry Cask Storage System Licenses and Certificates of Compliance''), which is currently being updated to include information regarding AMPs, among other things. The comments stated that the approval of this CoC system, ''should be put on hold until after the revised NUREG–1927 is final and any appropriate aging management issues are addressed in this CoC.''

The comments questioned some specific example AMPs discussed at public meetings, including questions regarding an example AMP for Chloride-Induced Stress Corrosion Cracking Tests (seismic concerns and sampling size), as well as the absence of an AMP given issues with damaged fuels and the ''unknowns of extended storage with high burnup fuel.'' In sum, these commenters felt that approval of CoCs, such as this one, should await the formulation and approval of aging management programs.

Response

These comments are outside the scope of this rulemaking which is limited to amending the spent fuel storage regulations by adding the UMAX Canister Storage System, CoC No. 1040, to the ''List of approved spent fuel storage casks'' in 10 CFR 72.214. This rulemaking is not making any changes to the regulations governing the standards for approval of a CoC.

The CoC for the HI–STORM UMAX is being issued for 20 years in accordance with 10 CFR part 72. According to the NRC staff's SER published in the **Federal Register** under Docket ID NRC–2014–0120, the staff has determined that the use of the HI–STORM UMAX Canister Storage System will be conducted in compliance with the applicable regulations of 10 CFR part 72, and the CoC should be approved for the initial 20-year term. There are currently no technical or regulatory requirements for the inclusion of AMPs for the initial 20-year CoC term. AMPs are required for spent fuel storage cask renewal which allows storage beyond 20 years, as provided in 10 CFR 72.240. The current regulatory requirements provide the necessary defense in depth for safe storage of spent nuclear fuel for at least 20 years.

Based on the regulations in 10 CFR part 72, an AMP will be required to be included in any renewal application for the HI–STORM UMAX Canister Storage System, for a duration beyond the initial 20-year term. The renewal application, if filed, will be required to comply with the applicable regulations, and consider applicable NRC aging management guidance available at the time of submittal. While NUREG–1927 may prove useful to applicants seeking to renew a CoC, because it does not provide guidance regarding applications seeking initial approval of certificates, there is no reason to await the guidance before proceeding with the addition of this system to the 10 CFR part 72 regulations.

Inspection Access

Several comments also questioned the ability of the underground storage system to be adequately inspected and potentially repaired if necessary during the initial certification period of 20 years, especially if the system was being used in a coastal environment where stress corrosion cracking could be an issue.

Response

The NRC is treating this comment as a significant adverse comment warranting clarification of the record. The NRC has evaluated the design of the HI–STORM UMAX Canister Storage System and has determined that the design is robust, and contains numbers of layers of acceptable confinement systems in compliance with 10 CFR part

72 requirements. In addition, the staff is not aware of empirical evidence that supports a finding that surveillance would be required in the initial certification period of the proposed CoC. This evaluation is documented in the NRC staff's SER under Docket ID NRC–2014–0120.

Furthermore, the NRC has evaluated the susceptibility to and effects of stress corrosion cracking and other corrosion mechanisms on safety significant systems for SNF dry cask storage (DCS) systems during an initial certification period. The staff has determined that the HI–STORM UMAX Canister Storage System, when used within the requirements of the proposed CoC, will safely store SNF and prevent radiation releases and exposure consistent with regulatory requirements.

Seismic Protection

Several comments also raised concerns regarding the ability of this CoC system to withstand seismic events, particularly if the system were to be used at specific sites with known seismic activity, such as San Onofre Nuclear Generating Station (SONGS).

Response

The NRC is treating this comment as a significant adverse comment warranting clarification of the record. This rulemaking would add a CoC system to the list of approved spent fuel storage casks in 10 CFR 72.214. The certification provided by this approval does not, in and of itself, authorize use of this system at any specific site. Instead, general licensees (a power reactor that stores spent fuel under a general Part 72 license) that wish to use this system must first ensure that other applicable requirements are met. (See 10 CFR 72.212).

The seismic design levels of the HI–STORM UMAX Canister Storage System as provided in this CoC are acceptable for most areas in the continental U.S. For locations that have potential seismic activity beyond those analyzed for this system, additional evaluations and certifications may be required before the system may be used in those locations. The NRC is currently evaluating an amendment request to the HI–STORM UMAX Canister Storage System that provides additional analysis intended to ensure the system's integrity during an earthquake with higher seismic demands, including the seismic demands at the location of SONGS. If the NRC approves that amendment request, the amended system could be selected for use at SONGS, provided regulatory requirements are met.

Bankruptcy

A comment also raised questions about the implications of the potential bankruptcy of corporations that seek CoC approvals.

Response

This comment is outside the scope of this rulemaking. This rulemaking would add a certified system to the list of spent fuel systems in 10 CFR 72.214 and does not seek to alter the standards for approval of a CoC system. In any event, NRC regulations in 10 CFR part 72 address the financial viability of licensees to ensure spent fuel management and decommissioning are funded. Pursuant to NRC requirements, once a general licensee accepts delivery of a storage system authorized by a CoC, the financial responsibility for maintaining and decommissioning the system become the responsibility of the general licensee (see 10 CFR 72.30(b), (c), (d), (e), and (f)).

Flood Protection

One comment stated that the design basis of the Watts Bar 2 reactor (not yet licensed for operation) intends that safe shut down could occur if there were a flood event that delivered 13½ feet of water at the reactor buildings. This comment raised the concern that the cask waste storage in an adjacent area would have equal or greater flooding.

Response

This rulemaking is limited to the approval of a CoC system to be added to the list of spent fuel storage casks in 10 CFR 72.214. This rulemaking does not propose any change to the standards for approval of a CoC, or the requirements that govern the use of this CoC by a general licensee. Therefore, this comment is outside the scope of this rulemaking.

The NRC's regulations at 10 CFR 72.212, ''Conditions of a general license issued under 10 CFR 72.210,'' require that a general licensee (a power reactor that stores spent fuel under a general part 72 license) perform written evaluations to ensure that the DCS systems used at the location meet the technical requirements of the CoC. The NRC inspects these evaluations prior to the first use of the DCS system and every three years after first use to ensure compliance with the terms of the CoC. If the CoC does not allow for water intrusion, then the general licensee is required to provide engineered measures to ensure that this condition does not occur.

High Burnup Fuel

Several comments also raised questions regarding the long-term acceptability of the extended storage of high burnup fuel (HBF).

Response

Most of the comments raising HBF as an issue did so in the context of the need for AMPs for approval of the CoC for the first 20 years, and that is beyond the scope of this rulemaking, as explained above.

To the extent commenters raised issues about the storage of HBF in the CoC for the first 20 years, the NRC is treating this portion of the comment as a significant adverse comment warranting clarification of the record. The NRC has evaluated the acceptability of storage of HBF for the initial 20-year certification term for the HI–STORM UMAX Canister Storage System. As documented in the NRC staff's SER under Docket ID NRC–2014–0120, the staff has determined that the use of the HI–STORM UMAX Canister Storage System, including storage of HBF, will be conducted in compliance with the applicable regulations of 10 CFR part 72, and the CoC should be approved for the initial 20-year term.

Storage beyond the initial term of 20 years will require the applicant to submit a license renewal application with the inclusion of AMPs addressing HBF. In that regard, a demonstration project is being planned by the U.S. Department of Energy to provide confirmatory data on the performance of HBF in DCS. The NRC plans to evaluate the data obtained from the project to confirm the accuracy of current models that are relied upon for authorizing the storage of HBF for extended storage periods beyond the initial 20-year certification term.

Duration of Certificate

Some comments also raised issues with the limited duration of this initial CoC for a term of only 20 years and stated that the systems should have to demonstrate safe storage of nuclear fuel for a much longer storage period.

Response

The issues of long-term storage and disposal of SNF are outside the scope of this CoC rulemaking. This rule is limited to the addition of this storage system to the list of approved designs in 10 CFR 72.214. The regulations governing the length of the CoC term are not within the changes proposed by this rule.

Inspector General's Report

One comment highlighted issues addressed in the 2014 NRC Inspector General's report of the SONGS steam generator replacement, entitled, ''NRC Oversight of Licensee's Use of 10 CFR 50.59 Process to Replace SONG'S Steam Generators (Case No. 13–006).''

Response

The issues raised by the NRC's IG report of the SONGS steam generator replacement are outside the scope of this rulemaking. This report is applicable only to that proposed steam generator replacement effort, and does not apply to nor is it related to this specific CoC rulemaking. Approval of this CoC is based upon a safety and environmental review of this specific CoC design as submitted by the vendor. If power reactor licensees wish to use this system at their specific sites, they must first ensure other applicable regulatory requirements are met (see 10 CFR 72.212).

IV. Voluntary Consensus Standards

The National Technology Transfer and Advancement Act of 1995 (Pub. L. 104–113) requires that Federal agencies use technical standards that are developed or adopted by voluntary consensus standards bodies unless the use of such a standard is inconsistent with applicable law or otherwise impractical. In this final rule, the NRC will add the Holtec International HI–STORM UMAX Canister Storage System design to the listing in 10 CFR 72.214. This action does not constitute the establishment of a standard that contains generally applicable requirements.

V. Agreement State Compatibility

Under the ''Policy Statement on Adequacy and Compatibility of Agreement State Programs'' approved by the Commission on June 30, 1997, and published in the **Federal Register** on September 3, 1997 (62 FR 46517), this final rule is classified as Compatibility Category ''NRC.'' Compatibility is not required for Category ''NRC'' regulations. The NRC program elements in this category are those that relate directly to areas of regulation reserved to the NRC by the Atomic Energy Act of 1954, as amended, or the provisions of 10 CFR. Although an Agreement State may not adopt program elements reserved to the NRC, it may wish to inform its licensees of certain requirements via a mechanism that is consistent with the particular State's administrative procedure laws, but does not confer regulatory authority on the State.

VI. Plain Writing

The Plain Writing Act of 2010 (Pub. L. 111–274), requires Federal agencies to write documents in a clear, concise, and well-organized manner. The NRC has written this document to be consistent with the Plain Writing Act as well as the Presidential Memorandum ''Plain Language in Government Writing,'' published June 10, 1998 (63 FR 31883).

VII. Environmental Assessment and Finding of No Significant Environmental Impact

A. The Action

The action is to amend 10 CFR 72.214 to add the Holtec International HI–STORM UMAX Canister Storage System to the listing within the ''List of approved spent fuel storage casks'' as CoC No. 1040. Under the National Environmental Policy Act of 1969, as amended, and the NRC's regulations in subpart A of 10 CFR part 51, ''Environmental Protection Regulations for Domestic Licensing and Related Regulatory Functions,'' the NRC has determined that this rule, if adopted, would not be a major Federal action significantly affecting the quality of the human environment and, therefore, an environmental impact statement is not required. The NRC has made a finding of no significant impact on the basis of this environmental assessment.

B. The Need for the Action

This final rule adds CoC No. 1040 for the Holtec International HI–STORM UMAX Canister Storage System design within the list of approved spent fuel storage casks that power reactor licensees can use to store spent fuel at reactor sites under a general license. Specifically, Holtec International's intent with this design is to provide an underground storage option compatible with the Holtec International HI–STORM FW System.

C. Environmental Impacts of the Action

On July 18, 1990 (55 FR 29181), the NRC issued an amendment to 10 CFR part 72 to provide for the storage of spent fuel under a general license in cask designs approved by the NRC. The potential environmental impact of using NRC-approved storage casks was initially analyzed in the environmental assessment for the 1990 final rule. The environmental assessment for this CoC addition tiers off of the environmental assessment for the July 18, 1990, final rule. Tiering on past environmental assessments is a standard process under the National Environmental Policy Act.

Holtec International HI–STORM UMAX Canister Storage Systems are designed to mitigate the effects of design basis accidents that could occur during storage. Design basis accidents account for human-induced events and the most severe natural phenomena reported for the site and surrounding area. Postulated accidents analyzed for an ISFSI, the type of facility at which a holder of a power reactor operating license would store spent fuel in casks in accordance with 10 CFR part 72, include tornado winds and tornado-generated missiles, a design basis earthquake, a design basis flood, an accidental cask drop, lightning effects, fire, explosions, and other incidents.

Considering the specific design requirements for each accident condition, the design of the HI–STORM UMAX Canister Storage System would prevent loss of containment, shielding, and criticality control. If there is no loss of containment, shielding, or criticality control, the environmental impacts would be insignificant. In addition, any resulting occupational exposure or offsite dose rates from the use of the HI–STORM UMAX Canister Storage System would remain well within the 10 CFR part 20 limits. Therefore, the proposed addition of CoC No. 1040 will not result in radiological or non-radiological environmental impacts that significantly differ from the environmental impacts evaluated in the environmental assessment supporting the July 18, 1990, final rule. There will be no significant change in the types or significant revisions in the amounts of effluent released, no significant increase in the individual or cumulative radiation exposure, and no significant increase in the potential for or consequences from radiological accidents. The staff documented its safety findings for this review in the SER.

D. Alternative to the Action

The alternative to this action is to withhold approval of this new design and issue a site-specific license to each utility that proposes to use the casks. This alternative would cost both the NRC and utilities more time and money for each site-specific license. Conducting site-specific reviews would ignore the procedures and criteria currently in place for the addition of new cask designs that can be used under a general license, and would be in conflict with NWPA direction to the Commission to approve technologies for the use of spent fuel storage at the sites of civilian nuclear power reactors without, to the maximum extent practicable, the need for additional site reviews. This alternative also would

tend to exclude new vendors from the business market without cause and would arbitrarily limit the choice of cask designs available to power reactor licensees. This final rule will eliminate the above problems and is consistent with previous Commission actions. Further, the rule will have no adverse effect on public health and safety. Therefore, the environmental impacts would be the same or less than the action.

E. Alternative Use of Resources

Approval of the addition of CoC No. 1040 would result in no irreversible commitments of resources.

F. Agencies and Persons Contacted

No agencies or persons outside the NRC were contacted in connection with the preparation of this environmental assessment.

G. Finding of No Significant Impact

The environmental impacts of the action have been reviewed under the requirements in 10 CFR part 51. Based on the foregoing environmental assessment, the NRC concludes that this final rule entitled, ''List of Approved Spent Fuel Storage Casks: Holtec International HI–STORM UMAX Canister Storage System, Certificate of Compliance No. 1040,'' will not have a significant effect on the human environment. Therefore, the NRC has determined that an environmental impact statement is not necessary for this final rule.

VIII. Paperwork Reduction Act Statement

This rule does not contain any information collection requirements and, therefore, is not subject to the requirements of the Paperwork Reduction Act of 1995 (44 U.S.C. 3501 *et seq.*).

Public Protection Notification

The NRC may not conduct or sponsor, and a person is not required to respond to, a request for information or an information collection requirement unless the requesting document displays a current valid OMB control number.

IX. Regulatory Analysis

On July 18, 1990 (55 FR 29181), the NRC issued an amendment to 10 CFR part 72 to provide for the storage of spent nuclear fuel under a general license in cask designs approved by the NRC. Any nuclear power reactor licensee can use NRC-approved cask designs to store spent nuclear fuel if it notifies the NRC in advance, the spent fuel is stored under the conditions specified in the cask's CoC, and the conditions of the general license are met. A list of NRC-approved cask designs is contained in 10 CFR 72.214.

By letter dated June 29, 2012, and as supplemented on July 16 and November 20, 2012; January 30, April 2, April 19, June 21, August 28, December 6, and December 31, 2013; and January 13, and January 28, 2014, Holtec International submitted an application to add the HI-STORM UMAX Canister Storage System.

The alternative to this action is to withhold approval of this new design and issue a site-specific license to each utility that proposes to use the casks. This alternative would cost both the NRC and utilities more time and money for each site-specific license. Conducting site-specific reviews would ignore the procedures and criteria currently in place for the addition of new cask designs that can be used under a general license, and would be in conflict with NWPA direction to the Commission to approve technologies for the use of spent fuel storage at the sites of civilian nuclear power reactors without, to the maximum extent practicable, the need for additional site reviews. This alternative also would tend to exclude new vendors from the business market without cause and would arbitrarily limit the choice of cask designs available to power reactor licensees. This final rule will eliminate the above problems and is consistent with previous Commission actions. Further, the rule will have no adverse effect on public health and safety.

Approval of this final rule is consistent with previous NRC actions. Further, as documented in the SER and the environmental assessment, the final rule will have no adverse effect on public health and safety or the environment. This final rule has no significant identifiable impact or benefit on other Government agencies. Based on this regulatory analysis, the NRC concludes that the requirements of the final rule are commensurate with the NRC's responsibilities for public health and safety and the common defense and security. No other available alternative is believed to be as satisfactory, and therefore, this action is recommended.

X. Regulatory Flexibility Certification

Under the Regulatory Flexibility Act of 1980 (5 U.S.C. 605(b)), the NRC certifies that this rule will not, if issued, have a significant economic impact on a substantial number of small entities. This final rule affects only nuclear power plant licensees and Holtec International. These entities do not fall within the scope of the definition of small entities set forth in the Regulatory Flexibility Act or the size standards established by the NRC (10 CFR 2.810).

XI. Backfitting and Issue Finality

The NRC has determined that the backfit rule (10 CFR 72.62) does not apply to this final rule. Therefore, a backfit analysis is not required. This final rule adds CoC No. 1040 for the Holtec International HI–STORM UMAX Canister Storage System to the ''List of approved spent fuel storage casks.''

The addition of CoC No. 1040 for the Holtec International HI–STORM UMAX Canister Storage System was initiated by Holtec International and was not submitted in response to new NRC requirements, or in response to an NRC request. The addition of CoC No. 1040 does not constitute backfitting under 10 CFR 72.62, 10 CFR 50.109(a)(1), or otherwise represent an inconsistency with the issue finality provisions applicable to combined licenses in 10 CFR part 52. Accordingly, no backfit analysis or additional documentation addressing the issue finality criteria in 10 CFR part 52 has been prepared by the staff.

XII. Congressional Review Act

In accordance with the Congressional Review Act of 1996 (5 U.S.C. 801–808), the NRC has determined that this action is not a rule as defined in the Congressional Review Act.

XIII. Availability of Documents

The documents identified in the following table are available to interested persons through one or more of the following methods, as indicated.

Document	ADAMS Accession No.
CoC No. 1040	ML14122A443
Safety Evaluation Report	ML14122A441
Technical Specifications, Appendix A	ML14122A444
Technical Specifications, Appendix B	ML14122A442

Document	ADAMS Accession No.
Application	ML121880102
Application supplemental July 16, 2012	ML12205A134
Application supplemental November 20, 2012	ML12348A483
Application supplemental January 30, 2013	ML13032A008
Application supplemental April 2, 2013	ML13107B249
Application supplemental April 19, 2013	ML13114A191
Application supplemental June 21, 2013	ML13175A363
Application supplemental August 28, 2013	ML13261A062
Application supplemental December 6, 2013	ML13343A169
Application supplemental December 31, 2013	ML14002A402
Application supplemental January 13, 2014	ML14015A145
Application supplemental January 28, 2014	ML14030A055
HI–STORM FW System FSAR	ML12363A284
HI–STORM UMAX Canister Storage System FSAR	ML12363A282

The NRC may post materials related to this document, including public comments, on the Federal rulemaking Web site at *http://www.regulations.gov* under Docket ID NRC–2014–0120. The Federal rulemaking Web site allows you to receive alerts when changes or additions occur in a docket folder. To subscribe: (1) Navigate to the docket folder (NRC–2014–0120); (2) click the ''Sign up for Email Alerts'' link; and (3) enter your email address and select how frequently you would like to receive emails (daily, weekly, or monthly).

List of Subjects in 10 CFR Part 72

Administrative practice and procedure, Criminal penalties, Manpower training programs, Nuclear materials, Occupational safety and health, Penalties, Radiation protection, Reporting and recordkeeping requirements, Security measures, Spent fuel, Whistleblowing.

For the reasons set out in the preamble and under the authority of the Atomic Energy Act of 1954, as amended; the Energy Reorganization Act of 1974, as amended; and 5 U.S.C. 552 and 553, the NRC is adopting the following amendments to 10 CFR part 72.

PART 72—LICENSING REQUIREMENTS FOR THE INDEPENDENT STORAGE OF SPENT NUCLEAR FUEL, HIGH-LEVEL RADIOACTIVE WASTE AND REACTOR-RELATED GREATER THAN CLASS C WASTE

■ 1. The authority citation for part 72 continues to read as follows:

Authority: Atomic Energy Act secs. 51, 53, 57, 62, 63, 65, 69, 81, 161, 182, 183, 184, 186, 187, 189, 223, 234, 274 (42 U.S.C. 2071, 2073, 2077, 2092, 2093, 2095, 2099, 2111, 2201, 2232, 2233, 2234, 2236, 2237, 2239, 2273, 2282, 2021); Energy Reorganization Act secs. 201, 202, 206, 211 (42 U.S.C. 5841, 5842, 5846, 5851); National Environmental Policy Act sec. 102 (42 U.S.C. 4332); Nuclear Waste Policy Act secs. 131, 132, 133, 135, 137, 141, 148 (42 U.S.C. 10151, 10152, 10153, 10155, 10157, 10161, 10168); Government Paperwork Elimination Act sec. 1704, (44 U.S.C. 3504 note); Energy Policy Act of 2005, Pub. L. 109–58, 119 Stat. 788 (2005).

Section 72.44(g) also issued under Nuclear Waste Policy Act secs. 142(b) and 148(c), (d) (42 U.S.C. 10162(b), 10168(c), (d)).

Section 72.46 also issued under Atomic Energy Act sec. 189 (42 U.S.C. 2239); Nuclear Waste Policy Act sec. 134 (42 U.S.C. 10154).

Section 72.96(d) also issued under Nuclear Waste Policy Act sec. 145(g) (42 U.S.C. 10165(g)).

Subpart J also issued under Nuclear Waste Policy Act secs. 117(a), 141(h) (42 U.S.C. 10137(a), 10161(h)).

Subpart K also issued under Nuclear Waste Policy Act sec. 218(a) (42 U.S.C. 10198).

■ 2. Section 72.214 is amended by adding Certificate of Compliance 1040 to read as follows:

§ 72.214 List of approved spent fuel storage casks.

* * * * *

Certificate Number: 1040.

Initial Certificate Effective Date: April 6, 2015.

SAR Submitted by: Holtec International, Inc.

SAR Title: Final Safety Analysis Report for the Holtec International HI–STORM UMAX Canister Storage System.

Docket Number: 72–1040.

Certificate Expiration Date: March 6, 2035.

Model Number: MPC–37, MPC–89.

Dated at Rockville, Maryland, this 24th day of February 2015.

For the Nuclear Regulatory Commission.

Mark A. Satorius,

Executive Director for Operations.

[FR Doc. 2015–05238 Filed 3–5–15; 8:45 am]

BILLING CODE 7590–01–P

DEPARTMENT OF ENERGY

10 CFR Part 431

[Docket Number EERE–2008–BT–STD–0015]

RIN 1904–AB86

Energy Conservation Program: Energy Conservation Standards for Walk-In Coolers and Freezers; Correction

AGENCY: Office of Energy Efficiency and Renewable Energy, Department of Energy.

ACTION: Final rule; correction.

SUMMARY: On June 3, 2014, the U.S. Department of Energy (DOE) issued a final rule adopting conservation standards for some classes of walk-in cooler and walk-in freezer components. The final rule was published with typographical errors to some of the reported values. DOE is providing corrections to address these errors. Neither the errors nor the corrections in this document affect the substance of the rulemaking or any of the conclusions reached in support of the final rule.

DATES: This correction is effective March 6, 2015.

FOR FURTHER INFORMATION CONTACT: Mr. John Cymbalsky, U.S. Department of Energy, Office of Energy Efficiency and Renewable Energy, Building Technologies Program, EE–5B, 1000 Independence Avenue SW., Washington, DC 20585–0121. Telephone: (202) 287–1692. Email: *walk-in_coolers_and_walk-in_freezers@EE.Doe.Gov.*

Mr. Michael Kido, U.S. Department of Energy, Office of the General Counsel, GC–33, 1000 Independence Avenue SW., Washington, DC 20585–0121. Telephone: (202) 586–8145. Email: *Michael.Kido@hq.doe.gov.*

SUPPLEMENTARY INFORMATION: The Department of Energy (''DOE'') is

U.S. NUCLEAR REGULATORY COMMISSION

NRC FORM 651
(10-2004)
10 CFR 72

CERTIFICATE OF COMPLIANCE FOR SPENT FUEL STORAGE CASKS

The U.S. Nuclear Regulatory Commission is issuing this Certificate of Compliance pursuant to Title 10 of the *Code of Federal Regulations*, Part 72, "Licensing Requirements for Independent Storage of Spent Nuclear Fuel and High-Level Radioactive Waste" (10 CFR Part 72). This certificate is issued in accordance with 10 CFR 72.238, certifying that the storage design and contents described below meet the applicable safety standards set forth in 10 CFR Part 72, Subpart L, and on the basis of the Final Safety Analysis Report (FSAR) of the cask design. This certificate is conditional upon fulfilling the requirements of 10 CFR Part 72, as applicable, and the conditions specified below.

Certificate No.	Effective Date	Expiration Date	Docket No.	Amendment No.	Amendment Effective Date	Package Identification No.
1040	April 6, 2015	April 6, 2035	72-1040	2	January 9, 2017	USA/72-1040

Issued To: (Name/Address)

Holtec International
Holtec Center
One Holtec Drive
Marlton, NJ 08053

Safety Analysis Report Title

Holtec International
Final Safety Analysis Report for the
HI-STORM FW MPC Storage System

Holtec International
Final Safety Analysis Report for the
HI-STORM UMAX Canister Storage System

This certificate is conditioned upon fulfilling the requirements of 10 CFR Part 72, as applicable, the attached Appendix A (Technical Specifications) and Appendix B (Approved Contents and Design Features), and the conditions specified below:

APPROVED SPENT FUEL STORAGE CASK

Model No.: HI-STORM UMAX Canister Storage System

DESCRIPTION:

The HI-STORM UMAX Canister Storage System consists of the following components: (1) interchangeable multi-purpose canisters (MPCs), which contain the fuel; (2) underground Vertical Ventilated Modules (VVMs), which contains the MPCs during storage; and (3) a transfer cask (HI-TRAC VW), which contains the MPC during loading, unloading and transfer operations. The MPC stores up to 37 pressurized water reactor fuel assemblies or up to 89 boiling water reactor fuel assemblies.

The HI-STORM UMAX Canister Storage System is certified as described in the" UMAX" Final Safety Analysis Report (FSAR) supplemented by the information on the MPCs and transfer cask in the HI-STORM FW FSAR, and in the U. S. Nuclear Regulatory Commission's (NRC) Safety Evaluation Report (SER) accompanying the Certificate of Compliance (CoC).

The MPC is the confinement system for the stored fuel. It is a welded, cylindrical canister with a honeycombed fuel basket, a baseplate, a lid, a closure ring, and the canister shell. All MPC components that may come into contact with spent fuel pool water or the ambient environment are made entirely of stainless steel or passivated aluminum/aluminum alloys. The canister shell, baseplate, lid, vent and drain port cover plates, and closure ring are the main confinement boundary components. All confinement boundary components are made entirely of stainless steel. The honeycombed basket provides criticality control.

DESCRIPTION (continued)

There are two types of MPCs permitted for storage in HI-STORM UMAX VVM: the MPC-37 and MPC-89. The number suffix indicates the maximum number of fuel assemblies permitted to be loaded in the MPC. Both MPC models have the same external diameter.

The HI-TRAC VW transfer cask provides shielding and structural protection of the MPC during loading, unloading, and movement of the MPC from the cask loading area to the VVM. The transfer cask is a multi-walled (carbon steel/lead/carbon steel) cylindrical vessel with a neutron shield jacket attached to the exterior and a retractable bottom lid used during transfer operations.

The HI-STORM UMAX VVM utilizes a storage design identified as an air-cooled vault or caisson. The HI-STORM UMAX VVM relies on vertical ventilation instead of conduction through the fill material around the VVM, as it is essentially a below-grade storage cavity. Air inlets and an air outlet allow air to circulate naturally through the cavity to cool the MPC inside. The subterranean steel structure is seal welded to prevent ingress of any groundwater in the MPC storage cavity from the surrounding subgrade, and it is mounted on a stiff foundation. The surrounding subgrade and a top surface pad provide significant radiation shielding. A loaded MPC is stored within the HI-STORM UMAX VVM in a vertical orientation.

HI-STORM UMAX Version MSE is a structurally strengthened embodiment of the VVM engineered for deployment at sites with its Design Basis Earthquake with ZPA in excess of 2.12Gs (resultant horizontal) and up to 1.0G (vertical).

CONDITIONS

1. OPERATING PROCEDURES

   Written operating procedures shall be prepared for handling, loading, movement, surveillance, and maintenance. The user's site-specific written operating procedures shall be consistent with the technical basis described in Chapter 9 of the FSAR.

2. ACCEPTANCE TESTS AND MAINTENANCE PROGRAM

   Written acceptance tests and a maintenance program shall be prepared consistent with the technical basis described in Chapter 10 of the FSAR. At completion of welding the MPC shell to baseplate, an MPC confinement weld helium leak test shall be performed using a helium mass spectrometer. This test shall include the base metals of the MPC shell and baseplate. A helium leakage test shall also be performed on the base metal of the fabricated MPC lid. The confinement boundary welds leakage rate test shall be performed in accordance with ANSI N14.5 to "leaktight" criterion. If a leakage rate exceeding the acceptance criteria is detected, then the area of leakage shall be determined and the area repaired per ASME Code Section III, Subsection NB, Article NB-4450 requirements. Re-testing shall be performed until the leakage rate acceptance criterion is met.

3. QUALITY ASSURANCE

   Activities in the areas of design, purchase, fabrication, assembly, inspection, testing, operation, maintenance, repair, modification of structures, systems and components, and decommissioning that are important-to-safety shall be conducted in accordance with a Commission-approved quality assurance program which satisfies the applicable requirements of 10 CFR Part 72, Subpart G, and which is established, maintained, and executed with regard to the storage system

4. HEAVY LOADS REQUIREMENTS

Each lift of an MPC or a HI-TRAC VW transfer cask must be made in accordance to the existing heavy loads requirements and procedures of the licensed facility at which the lift is made. A plant-specific review of the heavy load handling procedures (under 10 CFR 50.59 or 10 CFR 72.48, as applicable) is required to show operational compliance with existing plant specific heavy loads requirements. Lifting operations outside of structures governed by 10 CFR Part 50 must be in accordance with Section 5.2 of Appendix A.

5. APPROVED CONTENTS

Contents of the HI-STORM UMAX Canister Storage System must meet the fuel specifications given in Appendix B to this certificate.

6. DESIGN FEATURES

Features or characteristics for the site or system must be in accordance with Appendix B to this certificate.

7. CHANGES TO THE CERTIFICATE OF COMPLIANCE

The holder of this certificate who desires to make changes to the certificate, which includes Appendix A (Technical Specifications) and Appendix B (Approved Contents and Design Features), shall submit an application for amendment of the certificate.

8. PRE-OPERATIONAL TESTING AND TRAINING EXERCISE

A dry run training exercise of the loading, closure, handling, unloading, and transfer of the HI-STORM UMAX Canister Storage System shall be conducted by the licensee prior to the first use of the system to load spent fuel assemblies. The training exercise shall not be conducted with spent fuel in the MPC. The dry run may be performed in an alternate step sequence from the actual procedures, but all steps must be performed. The dry run shall include, but is not limited to the following:

a. Moving the MPC and the transfer cask into the spent fuel pool or cask loading pool.

b. Preparation of the HI-STORM UMAX Canister Storage System for fuel loading.

c. Selection and verification of specific fuel assemblies to ensure type conformance.

d. Loading specific assemblies and placing assemblies into the MPC (using a dummy fuel assembly), including appropriate independent verification.

e. Remote installation of the MPC lid and removal of the MPC and transfer cask from the spent fuel pool or cask loading pool.

f. MPC welding, NDE inspections, pressure testing, draining, moisture removal (by vacuum drying or forced helium dehydration, as applicable), and helium backfilling. (A mockup may be used for this dry-run exercise.)

g. Transfer of the MPC from the transfer cask to the VVM.

h. HI-STORM UMAX Canister Storage System unloading, including flooding MPC cavity and removing MPC lid welds. (A mockup may be used for this dry-run exercise.)

Any of the above steps can be omitted if the site has already successfully loaded a Holtec MPC System.

9. AUTHORIZATION

The HI-STORM UMAX Canister Storage System, which is authorized by this certificate, is hereby approved for general use by holders of 10 CFR Part 50 licenses for nuclear reactors at reactor sites under the general license issued pursuant to 10 CFR 72.210, subject to the conditions specified by 10 CFR 72.212, this certificate, and the attached Appendices A and B. The HI-STORM UMAX Canister Storage System may be fabricated and used in accordance with any approved amendment to CoC No. 1040 listed in 10 CFR 72.214. Each of the licensed HI-STORM UMAX Canister Storage System components (i.e., the MPC, overpack, and transfer cask), if fabricated in accordance with any of the approved CoC Amendments, may be used with one another provided an assessment is performed by the CoC holder that demonstrates design compatibility. .

FOR THE U. S. NUCLEAR REGULATORY COMMISSION

/RA/

John McKirgan, Chief
Spent Fuel Licensing Branch
Division of Spent Fuel Management
Office of Nuclear Material Safety
and Safeguards
Washington, DC 20555

Dated 1/6/17

Attachments:
1. Appendix A
2. Appendix B

CERTIFICATE OF SERVICE

I hereby certify that on July 20, 2020, I electronically filed the foregoing with the Clerk of the Court for the United States Court of Appeals for the Ninth Circuit by using the appellate CM/ECF system.

Participants in the case who are registered CM/ECF users will be served by the appellate CM/ECF system.

Dated: July 20, 2020

Respectfully submitted,
ALSTON & BIRD LLP

/ s/ Edward J. Casey

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No. 20-70899

IN THE UNITED STATES COURT OF APPEALS
FOR THE NINTH CIRCUIT

IN RE PUBLIC WATCHDOGS,

Petitioner,

v.

UNITED STATES NUCLEAR REGULATORY COMMISSION,

Respondent.

SOUTHERN CALIFORNIA EDISON COMPANY,
Intervenor.

INTERVENOR SOUTHERN CALIFORNIA EDISON'S SUPPLEMENTAL EXCERPTS OF RECORD

VOLUME 4 OF 8

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INTERVENOR SOUTHERN CALIFORNIA EDISON'S

SUPPLEMENTAL EXCERPTS OF RECORD:

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION IV
1600 EAST LAMAR BOULEVARD
ARLINGTON, TEXAS 76011-4511

July 9, 2019

EA-18-155

Mr. Doug Bauder
Vice President and Chief Nuclear Officer
Southern California Edison Company
San Onofre Nuclear Generating Station
P.O. Box 128
San Clemente, CA 92674-0128

SUBJECT: NRC SUPPLEMENTAL INSPECTION REPORT 050-00206/2018-006, 050-00361/2018-006, 050-00362/2018-006, 072-00041/2018-002

Dear Mr. Bauder:

This letter refers to a supplemental inspection using the U.S. Nuclear Regulatory Commission's (NRC's) Inspection Procedure 92702, "Follow-up on Traditional Enforcement Actions," conducted on January 28 through February 1, February 11-15, March 19, March 21-23, and April 10-13, 2019, at your facility in San Clemente, California. The inspection continued with in-office reviews of information provided by your staff from November 2018 through May 17, 2019.

The NRC performed this inspection to review corrective actions taken by the Southern California Edison Company in response to the misalignment of a loaded spent fuel storage canister as it was being downloaded into a storage vault at San Onofre Nuclear Generating Station (SONGS). Our initial review of the incident was documented in NRC Special Inspection Report 050-00206/2018-005, 050-00361/2018-005, 050-00362/2018-005, 072-00041/2018-001 and Notice of Violation (NRC's Agencywide Documents Access and Management System (ADAMS) Accession ML18341A172) and finalized in NRC letter "Notice of Violation and Proposed Imposition of Civil Penalty - $116,000 and NRC Inspection Report 050-00206/2018-005, 050-00361/2018-005, 050-00362/2018-005, 072-00041/2018 001," (ADAMS Accession ML19080A208).

The enclosed report documents the results of the supplemental inspection. The inspectors discussed the preliminary inspection findings with you and members of your staff on February 15, 2019, at the conclusion of a portion of the onsite inspection. A final exit briefing was conducted telephonically with Mr. Al Bates, Regulatory and Oversight Manager, and members of your staff on June 13, 2019.

The NRC performed this supplemental inspection to determine if: (1) the root and contributing causes of the significant performance issues were understood, (2) the extent of condition and extent of cause for the significant performance issues were identified, (3) the corrective actions taken to address and preclude repetition of significant performance issues were prompt and effective, and (4) the corrective action plans direct prompt actions to effectively address and

preclude repetition of significant performance issues. Additionally, the inspection team reviewed and determined if follow-up items from the NRC Special Inspection had been completed.

The NRC determined that your staff's causal evaluations to address the previously issued violations were adequately performed to the depth and breadth required. The NRC noted that your staff's evaluations identified that the primary root cause of the Severity Level II violation for failure to provide redundant lift protection features during downloading operations was that management failed to recognize the complexity and risks associated with a long duration fuel transfer campaign using a relatively new system design. Your staff determined that the primary cause for the Severity Level III violation for failure to make a report to the NRC was that management failed to recognize the required integration and application of 10 CFR Part 72 reporting requirements.

The NRC determined that your staff identified and implemented appropriate corrective actions to revise loading procedures, revise the reportability program, utilize equipment enhancements, require adequate training, enhance oversight of operations, and enhance the corrective action program at SONGS. The NRC also determined that your staff's extent of condition and extent of cause evaluations adequately reviewed whether other operations were susceptible to similar performance deficiencies. However, even though your causal evaluations and corrective actions were comprehensive, the NRC staff identified four observations associated with the evaluations and corrective actions.

Based on the results of the supplemental inspection, the NRC identified five findings that were identified as violations of NRC requirements and were determined to be Severity Level IV violations of low safety significance under the traditional enforcement process. The NRC is treating these violations as non-cited violations (NCVs) consistent with Section 2.3.2.a of the NRC Enforcement Policy.

If you contest the violations or significance of the NCVs, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-0001, with copies to: (1) the Regional Administrator, Region IV, and (2) the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001.

In accordance with 10 CFR 2.390 of the NRC's "Agency Rules of Practice and Procedure," a copy of this letter, its enclosure, and your response if you choose to provide one, will be made available electronically for public inspection in the NRC Public Document Room or from ADAMS. ADAMS is accessible from the NRC's Website at http://www.nrc.gov/reading-rm/adams.html. To the extent possible, your response should not include any personal privacy or proprietary information so that it can be made available to the Public without redaction.

If you have any questions regarding this inspection report, please contact Lee Brookhart at 817-200-1549, or the undersigned at 817-200-1223.

Sincerely,

Greg Warnick, Chief
Reactor Inspection Branch
Division of Nuclear Materials Safety

Docket Nos.: 050-00206; 050-00361;
050-00362; 072-00041
License Nos.: DPR-13; NPF-10; NPF-15

Enclosure:
Supplemental Inspection Report
050-00206/2018-006; 50-00361/2018-006;
050-00362/2018-006; 072-00041/2018-002
w/Attachments:
1. Supplemental Inspection Information
2. Radiological Surveys of ISFSI pads

U.S. NUCLEAR REGULATORY COMMISSION

REGION IV

Docket Nos.:	50-206; 50-361; 50-362; 72-041
License Nos.:	DPR-13; NPF-10; NPF-15
Report No.:	050-00206/2018-006; 050-00361/2018-006; 050-00362/2018-006; and 072-00041/2018-002
EA No.:	18-155
Licensee:	Southern California Edison Company
Facility:	San Onofre Nuclear Generating Station
Location:	San Clemente, CA 92674-012
Inspection Dates:	Onsite: January 28 - February 1, 2019; February 11 - 15, 2019; March 19, 21 - 23, 2019; and April 10 - 13, 2019 In-office review from November 2018 through May 17, 2019
Exit Meeting Date:	June 13, 2019
Inspectors:	Lee Brookhart, Senior ISFSI Inspector Reactor Inspection Branch Division of Nuclear Materials Safety, Region IV Eric Simpson, CHP, Health Physicist Reactor Inspection Branch Division of Nuclear Materials Safety, Region IV W. Chris Smith, Reactor/ISFSI Inspector Reactor Inspection Branch Division of Nuclear Materials Safety, Region IV Christopher Newport, Senior Resident Inspector Project Branch A, Diablo Canyon Division of Reactor Projects, Region IV
Accompanied by:	Janine F. Katanic, PhD, CHP, Acting Branch Chief Fuel Cycle and Decommissioning Branch Division of Nuclear Materials Safety, Region IV
Approved By:	Greg Warnick, Chief Reactor Inspection Branch Division of Nuclear Materials Safety, Region IV
Attachments:	1.) Supplemental Inspection Information 2.) Radiological Surveys of ISFSI Pads

EXECUTIVE SUMMARY

NRC Supplemental Inspection Report 050-00206/2018006; 050-00361/2018006; 050-00362/2018006; and 072-00041/2018-002

On January 28 through February 1; February 11-15; March 19; March 21-23; and April 10-13, 2019, the U.S. Nuclear Regulatory Commission performed an announced on-site Supplemental Inspection of the Independent Spent Fuel Storage Installation at the decommissioning San Onofre Nuclear Generating Station in San Clemente, California. The inspection continued with an in-office review of the licensee's analyses, procedures, and other materials gathered and provided prior to and after the on-site portion of the inspection through May 17, 2019.

The scope of the inspection was to evaluate and review the licensee's follow-up investigation, causal evaluations, implemented corrective actions, and planned corrective actions associated with violations described in the NRC's Special Inspection Report 050-00206/2018-005, 050-00361/2018-005, 050-00362/2018-005, and 072-00041/2018-001 and Notice of Violation (NRC's Agencywide Documents Access and Management System (ADAMS) Accession ML18341A172) and Notice of Violation and Proposed Imposition of Civil Penalty - $116,000 and NRC Inspection Report (ADAMS Accession ML19080A208).

The NRC determined that the licensee's causal evaluations were conducted to a level of detail commensurate with the significance of the problems and reached reasonable conclusions as to the root and contributing causes of the event. The NRC determined that completed or planned corrective actions were comprehensive and sufficient to address the performance issues that led to the previously identified violations.

Additionally, the inspectors identified five Severity Level IV, non-cited violations that involved failures to: (1) ensure appropriate quality standards on new equipment for downloading/withdrawal operations; (2) ensure purchased material conformed to the procurement documents for load sensing shackles; (3) ensure the loaded transfer cask and its conveyance was evaluated under the site-specific design basis earthquake; (4) provide adequate written basis for the initial 10 CFR 72.48 scratch evaluation; and (5) request the certificate holder to obtain a Certificate of Compliance amendment for use of the intermediate shelf in the spent fuel pool.

Follow-up on Traditional Enforcement Actions, Inspection Procedure 92702

- The inspectors independently reviewed the licensee's causal evaluations for the performance issues and significant findings that led to the August 3, 2018, misalignment incident. The NRC concluded that the evaluations were conducted to a level of detail commensurate with the significance of the problems and the root causes combined with the contributing causes adequately addressed the findings presented in the NRC Special Inspection Report. The inspectors also concluded that the root and contributing causes of the significant performance issues were understood by the licensee. One observation was identified by the NRC which related to the licensee's contributing causes. Subsequently, the licensee addressed and resolved the NRC observation by implementing additional corrective actions. (Section 1.2.1)

- The inspectors determined that the licensee evaluated the performance issues using systematic methodologies to identify root and contributing causes. The inspectors concluded that the licensee's causal evaluations addressed the extent of condition and extent of cause of the issues and appropriately considered safety culture traits. One observation was identified by the NRC regarding the licensee's extent of condition evaluation. Subsequently, the licensee addressed and resolved the issue by implementing additional corrective actions. (Section 1.2.2)

- The NRC concluded that the licensee's evaluations and corrective actions taken in the areas of licensee oversight, procedures, training, equipment, corrective action program, and reportability were appropriate to prevent recurrence of prior inspection findings and violations and were adequately prioritized with consideration to risk significance and regulatory compliance. The inspectors concluded that the licensee's completed corrective actions in the areas of training, corrective action program, and procedures were adequate to restore compliance and prevent recurrence for the relevant violations issued in the NRC Special Inspection Report, dated December 19, 2018. (Section 1.2.3.b (1)-(6))

- During the NRC's review, the inspectors identified two additional observations and two violations of NRC requirements relating to the licensee's corrective actions. The two violations were related to the licensee's failure to establish measures to ensure appropriate quality standards were specified in design documentation in accordance with 10 CFR 72.146 and the licensee's failure to establish measures to ensure that purchased equipment conformed to the procurement documents in accordance with 10 CFR 72.154 for the recent enhancements to fuel canister transfer equipment. The licensee entered the findings into the corrective action program as action requests 1218-20333 and 1219-52380. The violations were determined to have a low safety significance and the Severity Level IV violations were treated as non-cited violations. Subsequently, the licensee addressed and resolved the NRC observations and violations by implementing additional corrective actions. (Section 1.2.3.c)

- The inspectors evaluated and concluded that the licensee's corrective actions were prompt and effective, and the licensee had adequately established appropriate quantitative or qualitative measures of success for the actions implemented to monitor the effectiveness of the corrective actions to prevent recurrence. (Section 1.2.4)

<u>Follow-up of Events and Notices of Enforcement Discretion, Inspection Procedure 71153</u>

- The NRC reviewed Licensee Event Report 2018-001-1 (ADAMS Accession ML18317A060), dated November 8, 2018, for the licensee's actions which led to the inadvertent disablement of redundant important-to-safety slings during downloading operations on August 3, 2018. The NRC inspectors reviewed all the implemented and planned corrective actions and found them to be adequate to restore compliance and prevent recurrence. This licensee event report is closed. (Section 2.2.1)

- The NRC reviewed Licensee Event Report 2018-002-0 (ADAMS Accession ML19050A170), dated February 14, 2019. The licensee notified the

NRC that previous operations utilizing the low-profile-transporter were performed outside the clearance limits calculated in the station's site-specific seismic analysis. The NRC inspectors reviewed all the implemented corrective actions and found them to be adequate to restore compliance and prevent recurrence. The licensee event report described that an analysis was still in progress to determine if past operations were acceptable. This licensee event report remains open, pending NRC review of the additional information. (Section 2.2.2)

- The NRC reviewed Licensee Event Notification 53858, dated February 2, 2019. The licensee notified the NRC that previous operations utilizing the vertical cask transporter had been performed, for short periods of time, outside conditions described in the station's site-specific seismic analysis. Specifically, the licensee prematurely removed the seismic restraint band prior to stack-up operations. The NRC inspectors reviewed all the implemented and planned corrective actions and found them to be adequate to restore compliance and prevent recurrence. This licensee event notification is closed. (Section 2.2.3)

- The inspectors documented a violation of Certification of Compliance 72-1040, Appendix B, Technical Specification 3.4.15, for the licensee's failure to conduct transportation operations in accordance with the station's site-specific seismic analysis. Specifically, the NRC identified, the licensee prematurely removed the seismic restraint band prior to stack-up operations during vertical cask transporter operations. The licensee entered the finding into the corrective action program as action requests 0219-88442, 0219-22465, and 0319-95843. The NRC determined that the finding was of low safety significance since the licensee had re-performed the seismic evaluations restoring compliance and demonstrated the canister and its conveyance would not have tipped-over or slid off the haul route during those transportation operations. This Severity Level IV violation was treated as a non-cited violation. (Section 2.2.4)

- As a follow-up to the Special Inspection Charter, the NRC reviewed the licensee's evaluation to analyze the potential effects of dropping a canister approximately 18 feet onto the base of the UMAX vault. The NRC agreed with the evaluation conclusion that the canister would not have breached had the canister fell to the bottom of the UMAX vault. Additionally, the NRC concluded that the canister would have continued to perform all safety functions, including structural, thermal, criticality control, and shielding. (Section 2.2.5.a)

- The licensee performed a change under the 10 CFR 72.48 process to evaluate and accept scratches from incidental contact during insertion and withdrawal operations on previously loaded and future canisters placed in the UMAX independent spent fuel storage installation. The licensee's subsequent written evaluation, based on in-situ visual assessments and statistical analyses of eight loaded canisters, was adequate to demonstrate that the proposed change would not affect the canisters' ability to meet the confinement design function and structural functions as specified in the Holtec Final Safety Analysis Report.

  The licensee's evaluation also demonstrated that American Society of Mechanical Engineers Section III code tolerances for wear were met and did not require a change to the storage system's technical specifications. The NRC utilized the data

obtained through the visual assessments to perform independent statistical assessments using several models that were appropriate for the sample size. The NRC concluded that the conclusion presented by the Southern California Edison Company was conservative and reasonably bounded the maximum anticipated scratch or wear depth resulting from routine operational activities. The NRC concluded the licensee's 10 CFR 72.48 change did not require prior NRC review and approval through an amendment request. (Section 2.2.5.b)

- The inspection results documented one violation of NRC requirements for the licensee's failure to include an adequate evaluation to support a design change in accordance with 10 CFR 72.48. The NRC identified that the licensee's original evaluations to allow scratching and gouging on canisters contained multiple errors and inadequacies, and the NRC determined that the calculation could not adequately bound the maximum possible scratch depth on a canister.

  The licensee entered the finding into the corrective action program as action requests 1218-11302 and 0219-96601. The NRC determined that the finding was of low safety significance since the licensee re-performed the written evaluation utilizing in-situ visual assessment and statistical analyses that calculated a maximum probable scratch depth, which provided an adequate basis for the determination that the change did not require NRC review through an amendment request. This Severity Level IV violation was treated as a non-cited violation. (Section 2.2.6)

- The NRC closed an Unresolved Item from NRC Inspection Report 07200041/2017-001 dated, August 24, 2018 (ADAMS Accession ML18200A400). The Unresolved Item was related to a 10 CFR 72.48 evaluation for the scenario of a hypothetical accident of the loaded HI-TRAC VW transfer cask contacting the sides and bottom of the spent fuel pool during the short period of time that a loaded multi-purpose canister was in an unconstrained condition on an intermediate shelf in the spent fuel pool.

  The inspectors determined one violation of NRC requirements occurred, for the licensee's failure to request the certificate holder to obtain an amendment prior to implementing a change in accordance with 10 CFR 72.48. The licensee's design change created the possibility of an accident of a different type than any previously evaluated in the Holtec Final Safety Analysis Report. The licensee entered the issue into the corrective action program as action requests 0718-10512 and 0617-86918. The NRC determined that the finding was of low safety significance since the accident condition had been analyzed and NRC approved in NUREG-0712 "Safety Evaluation Report related to the operation of SONGS Units 2 and 3, dated February 1981," and described in the San Onofre Nuclear Generating Station Decommissioning Safety Analysis Report. The licensee restored compliance by revising the loading procedures to no longer utilize the intermediate shelf in the spent fuel pool. This Severity Level IV violation was treated as a non-cited violation. (Section 2.2.7)

- The inspection team observed the licensee perform several dry run exercises utilizing a simulated canister. On January 28, 2019, the licensee successfully demonstrated operations utilizing the low-profile transporter to transport the simulated canister within the transfer cask to the independent spent fuel storage

installation pad while maintaining compliance with the station's site-specific seismic analysis. On February 14, 2019, the licensee successfully demonstrated removal of the transfer cask from the bottom of the spent fuel pool directly to the cask washdown pit without utilizing the intermediate shelf in the spent fuel pool.

On January 28-30, 2019, the inspection team observed the licensee implementing all the corrective action enhancements to download and retrieve a simulated canister at the independent spent fuel storage installation pad. These exercises contained: (1) all vendor personnel trained and qualified under the new training program, (2) use of more personnel, located in strategic positions to observe canister downloading, (3) utilization of the enhanced procedures, (4) implementation of the new canister transfer monitoring equipment, and (5) enhanced oversight by licensee personnel qualified under a new oversight training program. The station was fully successful in downloading and retrieving the canister during the exercises and the corrective actions taken were determined by the inspectors to be adequate to restore compliance and prevent recurrence of the performance issues that led to the misalignment event. (Section 2.2.8)

- The NRC inspectors closed the violation for the licensee failure to ensure that redundant drop protection features were available during the August 3, 2018, misalignment event. The NRC thoroughly reviewed the licensee's completed and proposed corrective actions related to the misalignment event and concluded the corrective actions were adequate to restore compliance, address extent of condition, and prevent recurrence. (Section 2.2.9)

- The NRC inspectors performed independent measurements and verifications of the radiological conditions at the licensee's independent spent fuel storage installation. The inspectors measured various locations including background areas, public access areas, owner-controlled areas, and representative locations on both generally licensed independent spent fuel storage installation pads. Based on the number and age of canisters in service, the NRC did not identify any radiological concerns during the survey. Additionally, the NRC did not identify any measurements at the owner-controlled area boundary or in the public access areas to be above normal background measurements. (Section 2.2.10)

REPORT DETAILS

Summary of Plant Activities

The San Onofre Nuclear Generating Station (SONGS) independent spent fuel storage installation (ISFSI) consists of two ISFSI designs located adjacent to each other. The Transnuclear, Inc. (TN) nuclear horizontal modular storage (NUHOMS) ISFSI contains 51 loaded concrete advanced horizontal storage modules (AHSMs), which hold stainless steel dry shielded canisters (DSCs). Spent fuel from all three reactors are stored at the NUHOMS ISFSI in 50 of the storage modules.

Greater-than-Class-C (GTCC) waste from the Unit 1 reactor decommissioning project is stored in one module. There is a total of 63 AHSMs on the NUHOMS ISFSI pad. The 12 empty AHSMs will be available for storage of additional GTCC waste from Units 2 and 3. The 63 AHSMs currently on the pad are designed for the 24PT1-DSC (Unit 1 fuel) and 24PT4-DSC (Unit 2/3 fuel) canisters, which hold a maximum of 24 spent fuel assemblies. The 24PT1-DSCs are loaded and maintained under Amendment 0 of Certificate of Compliance (CoC) No. 72-1029 and the 24PT4-DSCs are loaded and maintained under Amendment 1 of CoC No. 72-1029. Both systems were being maintained under Final Safety Analysis Report (FSAR), Revision 5.

The Holtec UMAX ISFSI portion was designed to hold 75 multi-purpose canisters (MPCs). The MPC-37s contain 37 pressurized water reactor fuel assemblies in accordance with UMAX CoC No. 72-1040, Amendment 2, the HI-STORM UMAX FSAR, Revision 4, and the HI-STORM FW FSAR, Revision 5. The licensee has 29 loaded canisters in service at the UMAX ISFSI. A 30th canister had been loaded, welded, dried, and helium backfilled, but remained inside the Unit 3 spent fuel building. The licensee ceased all loading operations to address the investigation and implementation of corrective actions associated with the August 3, 2018, misalignment incident.

1 Followup on Traditional Enforcement Actions (Inspection Procedure 92702)

1.1 <u>Inspection Scope</u>

The NRC performed this supplemental inspection in accordance with Inspection Procedure 92702, "Follow-up of Traditional Enforcement Actions Including Violations, Deviations, Confirmatory Action Letters, Confirmatory Orders, and Alternative Dispute Resolution Confirmatory Orders," to assess the licensee's response to the issues identified during the inspection documented in NRC Special Inspection Report dated, December 19, 2018, "Special Inspection Report 050-00206/2018-005, 050-00361/2018-005, 050-00362/2018-005, and 072-00041/2018-001 and Notice of Violation," (NRC Special Inspection) (ADAMS Accession ML18341A172), using the following inspection objectives:

- Objective 1: To assure that the root and contributing causes of significant performance issues were understood;
- Objective 2: To independently assess and assure that the extent of condition and extent of cause of significant performance issues were identified;

- Objective 3: To assure that corrective actions taken to address and preclude repetition of significant performance issues were prompt and effective;
- Objective 4: To assure that corrective action plans directed prompt actions to effectively address and preclude repetition of significant performance issues.

The NRC Special Inspection Report documented the NRC's review of an August 3, 2018, misalignment incident that occurred when a loaded spent fuel canister came to rest on the shield ring near the top of the UMAX ISFSI vault, which prevented it from being fully lowered into the storage vault. At that time, the important-to-safety (ITS) rigging and lifting slings were slack and were no longer capable of performing their safety function of supporting and controlling the loaded canister. This failure to maintain redundant drop protection placed the canister (No. 29) in an unanalyzed condition because the ISFSI FSAR assumed a postulated drop was a non-credible event. The estimated time the canister was in an unsupported position was approximately 45 minutes.

Following the misalignment incident, the licensee failed to notify the NRC that ITS equipment was disabled and would fail to function as designed when required by the Certificate of Compliance to provide redundant drop protection features to prevent and mitigate the consequences of a drop accident and no redundant equipment was available and operable to perform the required safety function. The licensee's failure to make the required report to the NRC existed for 39 days until the report was submitted and compliance restored.

On March 25, 2019, the NRC issued letter, "Notice of Violation and Proposed Imposition of Civil Penalty - $116,000 and NRC Inspection Report 050-00206/2018-005, 050-00361/2018-005, 050-00362/2018-005, 072-00041/2018-001," (ADAMS Accession ML19080A208), to document the final significance determination for the identified escalated violations. The licensee's failure to ensure ITS equipment was available to provide redundant drop protection during downloading operations was characterized as a finding having significant safety consequence and was identified as a Severity Level II violation of NRC requirements. The licensee's failure to make a timely notification to the NRC Headquarters Operations Center for the August 3, 2018, disabling of ITS equipment impacted the ability of the NRC to perform its regulatory oversight function and was identified as a Severity Level III violation of NRC requirements.

The inspectors reviewed the licensee's causal evaluations and supplemental information during the inspection period. The inspectors held discussions with licensee personnel to determine if the root causes, contributing causes, and the contribution of safety culture components related to the issues were understood, and that corrective actions taken or planned were appropriate to address the causes and preclude repetition.

1.2 Observations and Findings

1.2.1 Problem Identification and Cause Evaluations (Objective 1)

a. Overview

The inspectors verified that the licensee's evaluations adequately documented identification of the issues. The violation involving failure to provide redundant drop protection features during downloading operations was self-revealed and the violation for

failure to make a report to the NRC was NRC identified. The inspectors determined that the evaluations documented how long the issues existed and prior opportunities for identification. The inspectors also determined that the evaluation documented significant plant-specific consequences and compliance concerns associated with the issues.

The inspectors evaluated whether the licensee's causal evaluations were conducted to a level of detail commensurate with the significance of the problem, and whether the licensee's evaluations included consideration of prior occurrences of the problem and knowledge of prior operating experience.

b. Assessment

The licensee performed four causal evaluations to address the issues resulting from the August 3, 2018, misalignment incident. The four causal evaluations were tracked in the licensee's Corrective Action Program (CAP) and addressed the following areas:

- Root Cause Evaluation (RCE) Quality Investigation (QI)-2529 was initiated to identify the root causes and corrective actions necessary to address the misalignment event and enhance Holtec's processes and procedures to prevent recurrence.

- Apparent Cause Evaluation (ACE) (Action Report (AR) 0818-20356) was initiated to determine why the Southern California Edison Company (SCE) oversight was ineffective in preventing the misalignment event.

- Common Cause Evaluation (CCE) (AR 0618-77146) was initiated to identify common issues that challenged construction of ISFSI facilities and fuel transfer operations.

- Reportability Root-Cause Evaluation (RRCE) (AR 1218-33805) was initiated to determine why a report was not submitted to the NRC within the required time-frame.

The RCE QI-2529 identified one root cause and five contributing causes. Specifically, the evaluation determined that the root cause of this event was: "*Holtec Management failed to recognize the complexity and risks associated with fuel transfer operation while using a relatively new system design (UMAX) in conjunction with a long duration campaign, and thus, did not implement necessary program improvements or the necessary level of oversight.*" The licensee determined that the contributing causes were: (1) inadequate content in procedures to recognize special conditions related to a new equipment system (UMAX); (2) the design review process did not ensure that unintended consequences of design features were captured; (3) communication protocols with a chain of command established during canister movements were not well defined; (4) Holtec had not established a continuous learning environment which promoted the use of internal and external operating experience; (5) the Holtec Training Program did not consider the uniqueness of the UMAX system relative to the other HI-STORM systems nor the uniqueness of challenges raised in a long-term project, which led to not fully establishing qualification or proficiency requirements for the task performers.

As a result, Holtec identified and addressed a significant number of weaknesses in the areas of design review, procedures, training, safety culture, operating experience,

corrective action processes, and communications. The SCE reviewed and approved Holtec RCE QI-2529 and the associated corrective actions through the SONGS's Corrective Action Program (CAP) as Action Request (AR) 0818-76588.

The ACE 0818-20356 identified one apparent cause and two contributing causes. Specifically, the evaluation determined that the apparent cause was: *"SCE ISFSI Project Management failed to establish a rigorous process to ensure technically accurate Holtec Procedures, adequate SCE and Holtec training to support procedure implementation, and sufficiently detailed Oversight Specialist guidance."* The licensee determined that the contributing causes were: (1) SCE project management observations were not being routinely performed, and (2) SCE project management had not consistently reinforced initiation of an AR for deviations from what was expected, even if covered by procedure, or that result in additional dose. As a result, the licensee identified and addressed a significant number of weaknesses in the areas of vendor material reviews, training for oversight individuals, oversight processes, safety culture, operating experience, and corrective action processes.

The CCE 0618-77146 identified one common cause and one contributing cause. Specifically, the licensee's evaluation determined that the common cause was: *"Holtec did not staff the project with knowledgeable experienced personnel to effectively manage, and administer, the Holtec Quality Assurance Program or the Holtec Corrective Action Program."* The licensee determined that the contributing cause was: (1) Holtec procedures and processes that feed into the Holtec CAP, were not sufficiently detailed or prescriptive to guide or instruct a person with limited quality and CAP experience to identify and effectively resolve conditions adverse to quality and/or trends in a timely manner. As a result, the licensee identified and addressed weaknesses in the areas of CAP processes and CAP training in both the Holtec and SCE CAP programs.

The RRCE 1218-33805 identified one root cause and two contributing causes. Specifcally, the licensee's evaluation determined that the root cause was: *"SCE Management failed to recognize the transition to fuel transfer operations as requiring the integration, familiarization, and application of 10 CFR 72.75 reporting requirements into plant processes."* The licensee determined that the contributing causes were: (1) There was a lack of guidance to facilitate understanding of the wording in 10 CFR 72.75(d); and (2) SCE management did not encourage, and the organization did not demonstrate, a conservative bias for reporting. As a result, the licensee identified and addressed weaknesses in the areas of reportability training and the reportability process.

c. Observations

An observation was identified by the NRC inspectors during the review of the four causal evaluations, which related to contributing causes. The inspectors identified that the licensee failed to address one potential contributing cause of the spent fuel storage canister downloading event. Specifically, the inspectors noted that the site emphasis on minimizing radiation dose directly led to personnel critical to the oversight of the downloading evolution being relocated to a low dose area where direct observation of the downloading activities was not possible. This led to a partial loss of command and control of the evolution and was likely a contributing cause of the event.

The inspectors noted that this potential causal factor was identified in the ACE 0818-20356. However, the causal factor was not identified as a contributing factor

nor tracked as a specific corrective action in the ACE 0818-20356 or RCE QI-2529. The inspectors identified through interviews with the loading personnel that training on this causal factor was conducted for personnel involved in future downloading operations. However, the inspectors were unable to verify the subject was captured in the licensee's training lessons and training presentations. In response, the licensee initiated corrective action AR 0219-25489 to address the NRC identified issue. Corrective actions taken included revising the radiation protection work plan and training lesson plans to include radiation protection lessons learned. Corrective actions taken were adequate to resolve the NRC observation.

d. Conclusions

The inspectors independently reviewed the licensee's causal evaluations for the performance issues and significant findings that existed which led to the misalignment incident. The NRC concluded that the evaluations were conducted to a level of detail commensurate with the significance of the problems and the root causes combined with the contributing causes and adequately addressed the findings presented in the NRC Special Inspection Report. The inspectors also concluded that the root and contributing causes of the significant performance issues were understood by the licensee. One observation was identified by the NRC related to the identified contributing causes, which was subsequently entered into the CAP and addressed by the licensee to resolve the NRC concern. As a result, Inspection Objective 1 was met.

1.2.2 Extent of Condition and Extent of Cause Evaluation (Objective 2)

a. Overview

The inspectors verified that the significant performance issues were evaluated using a systematic methodology. The inspectors evaluated whether the root-cause evaluation was conducted to a level of detail commensurate with the significance of the problems, and that it included a consideration of prior occurrences of the problems and knowledge of prior operating experience. Additionally, the inspectors assessed whether the causal evaluations addressed the extent of condition and extent of cause associated with the significant performance issues and assessed whether the licensee appropriately considered safety culture traits.

b. Assessment

The inspectors determined that the licensee's causal evaluations used systematic methodologies and were conducted to a level of detail commensurate with the significance of the problems. The identified causes, discussed in the previous section, are the results of an aggregate review using multiple analytical techniques. The inspectors also determined that the causal evaluations included a consideration of prior occurrences of the problems and knowledge of prior operating experience.

The licensee used the following systematic methods to complete the four causal evaluations:

- The RCE QI-2529 applied: 1.) Five Whys Approach; 2.) Barrier Analysis; 3.) Organizational and Programmatic Assessment; 4.) Human Factor Analysis; 5.) Comparative Time Line; and 6.) Safety Culture Assessment

- The ACE 0818-20356 applied: 1.) Cause and Effect Charting; and 2.) Lines of Inquiry List
- The CCE 0618-77146 applied: 1.) Pareto Chart; and 2.) Bin Assessment
- The RRCE 1218-33805 applied: 1.) Cause and Effect Charting; 2.) Barrier Analysis; and 3.) Safety Culture Assessment

The inspectors determined whether the licensee's causal evaluations addressed extent of condition and extent of cause of the problems identified in the reviews. Specifically, the RCE QI-2529 assessed the degree that the actual condition may exist in plant equipment, processes, or human performance that could result in the same or similar consequences. The extent of cause-initiated changes within Holtec's processes, which included evaluation of other facility's downloading procedures, verification of crew composition, qualifications, lessons learned, training enhancements, and design reviews.

The licensee's ACE 0818-20356 assessed all other fuel movements and heavy lifts at SONGS. The extent of cause review-initiated changes in all other ISFSI loading procedures and reviews of ISFSI non-loading procedures. Additionally, changes were initiated in licensee oversight of other vendor activities, including decommissioning activities, in the areas of training, document reviews, oversight observation programs, and lessons learned.

The licensee's RRCE 1218-33805 assessed additional areas where reportability may have been required but was not made to the NRC. Through that review the licensee determined one notification to the NRC was required. This notification related to the lateral clearance between the low-profile transporter and other structures (e.g. light posts), and the low-profile transporter's center of gravity was not maintained in accordance with the seismically analyzed limits. The licensee made the required notification to the NRC under 10 CFR 72.75(d)(1) on December 20, 2018 (Event Notification (EN) 53798) (see Section 2.2.2 for further discussion of the licensee event report). The extent of cause review addressed other reporting requirements within 10 CFR 72.75 and other applicable federal regulations. Additional actions were taken to enhance training and procedural processes to ensure reporting requirements would be followed as required in 10 CFR Parts 20, 49, 50, 71, and 72.

c. Observations

An observation was identified by the inspectors during the extent of condition review for the four causal evaluations. The inspectors identified that the licensee failed to perform one of the extent of condition reviews described in ACE 0818-20356. Specifically, Corrective Action (CA) 17 (CA-17), which stated, for Holtec procedures, other than operating procedures, determine which ones have a potential impact on operations and conduct a review using the review guidance in Corrective Action to Prevent Recurrence 2 (CAPR-2). The CAPR-2 task actions were to include additional requirements in procedure S0123-XV-93, "Contractor Oversight," to ensure a more rigorous review was completed by SCE oversight staff before accepting the document for use at the station.

The NRC inspectors identified that this review of Holtec non-loading/maintenance procedures had not been performed as specified in CA-17. In response, the licensee initiated corrective action AR 0818-20356 to perform the required review. The review included approximately 15 Holtec procedures which involved areas of crane maintenance, special lifting device maintenance, vertical cask transporter (VCT) maintenance, foreign material control program, weld examination program, etc. The inspectors reviewed the comments and discrepancies that were identified by the SCE staff from the review. The documentation of the review included a table of all comments identified by SCE staff and the revised procedures that documented that identified issues were changed. The corrective actions taken were adequate to address the NRC observation.

d. Conclusions

The inspectors determined that the licensee evaluated the issues using systematic methodologies to identify root and contributing causes. Additionally, the inspectors concluded that the licensee's causal evaluations addressed the extent of condition and extent of cause of the issues and appropriately considered safety culture traits. One observation was identified by the inspectors which was related to the extent of condition review. The licensee addressed the issue by taking adequate corrective actions. As a result, Objective 2 was met.

1.2.3 Corrective Actions Taken (Objective 3)

a. Overview

The inspectors reviewed the licensee's causal evaluations to assess whether appropriate corrective actions were specified for the root and contributing causes or that the licensee had an adequate evaluation for why no corrective actions were necessary. The inspectors also assessed whether the corrective actions had been prioritized with consideration of the safety significance and regulatory compliance. The inspectors evaluated whether the corrective actions taken to address and preclude repetition of significant performance issues were prompt and effective, and whether the violations, related to the NRC Special Inspection, had been adequately addressed.

b. Assessment

The corrective actions taken by the licensee are described below in the following areas: (1) Licensee Oversight; (2) Procedures; (3) Training; (4) Equipment and Personnel; (5) Corrective Action Program; and (6) Reportability.

(1) Licensee Oversight

The licensee's ACE 0818-20356, contained the majority of the corrective actions for the area of licensee oversight. Corrective actions drove extensive changes to the training and qualification program that an ISFSI oversight specialist is required to complete. The licensee increased the number of oversight specialists that directly observe ISFSI operations from approximately 10 to 14 individuals. All existing and new specialists were required to complete the enhanced qualification program requirements. The licensee assigned a specific training manager to oversee the enhanced training/qualification program. The licensee developed new lesson plans as

part of the qualification process. The new lesson plans included training on new load monitoring equipment, new task specific guides for field observations, new oversight roles and responsibilities, expectations, procedure changes, use of the corrective action program, acceptance review process changes, lessons learned, and other topics.

The licensee developed procedure G-XV93-PTP-01, "Pool to Pad Job Guide Desktop Guide," Revision 0. The inspectors reviewed the procedure and observed that it contained job guides for the ISFSI oversight specialists to use as a tool to assist in preparation and observational direction on the critical tasks during fuel transfer operations. The procedure described key elements of all work activities, detailing how and why tasks were critical. The guide directed the ISFSI oversight specialists to which specific tasks were required to be observed. The inspectors' review concluded that the task guide contained all critical tasks associated with fuel operations.

The licensee's site acceptance process of vendor procedures and training documents were revised. The changes included additional requirements to ensure a rigorous review prior to procedure acceptance and use at SONGS. The inspectors reviewed the procedure changes and the package of reviews conducted by oversight personnel to ensure all new and previously accepted documents received the same level of review. The inspectors concluded that the changes were appropriate, the reviews were thorough, and all identified issues were adequately addressed and corrected.

The licensee's changes included developing an oversight management organization to conduct observations on oversight specialists while they performed their field duties. The program included peer-to-peer observational requirements by decommissioning oversight personnel, as well as management observational requirements of the ISFSI oversight personnel. The program also contained effectiveness review requirements to ensure the required peer and management observations were effective and completed as required. The inspectors reviewed audit packages that were performed on oversight specialists during training exercises. The peer and management observations were well documented, and all identified enhancements and coaching items were captured in the licensee's CAP. The NRC concluded that the licensee had made substantial improvements throughout the ISFSI oversight program. No NRC observations were identified in this area.

(2) Canister Handling Procedures

The licensee's ACE 0818-20356 and RCE QI-2529 evaluations of the misalignment incident identified corrective actions which were intended to address procedural inadequacies that contributed to the incident. To address identified issues, the causal evaluations recommended corrective actions for the procedures that included the following changes: (1) continuous monitoring of weight sensing equipment during downloading operations; (2) establishment of clear underload criteria for when to halt downloading operations; (3) defining crew member roles and responsibilities by title; (4) listing qualification requirements for the specified roles; (5) listing critical steps in procedures; (6) defining responsibilities of cask loading supervisors; and (7) identifying areas where escalated management oversight was required.

Changes (1) and (2) were specifically directed at Holtec Procedure HPP-2464-400, "MPC Transfer at SONGS," Revision 17. The NRC inspectors reviewed the procedure revisions that included the new requirement to continuously monitor the canister

weight. The procedure revisions included establishment of clear underload criteria for when to halt downloading operations. The revised procedure directed the VCT operator and VCT platform rigger to maintain visual contact with the VCT control panel screen, load shackle tablet weight display screen, and downloader slings during canister downloading operations.

Procedure HPP-2464-400, Section 7.6, "Canister Download into Cavity Enclosure Container (CEC)," was revised to include steps to record the canister weight and to establish an underload restriction value. These changes included contingency steps for re-centering the canister if downloading operators noted a restriction in downward travel. The procedure also directed stop work requirements if certain underload conditions were experienced. Those actions included withdrawing the canister back into the transfer cask, making the appropriate notifications to site management, and condition report initiation into the CAP.

Changes (3) through (7) were applied to all operational procedures related to dry cask storage operations at SONGS. Those procedures included HPP-2464-100, "MPC Pre-Operation Inspection;" HPP-2464-200, "MPC Loading at SONGS;" HPP-2464-300, "MPC Sealing;" HPP-2464-400, "MPC Transfer at SONGS;" HPP-2464-500, "MPC Unloading;" and HPP-2464-600, "Responding to Abnormal Conditions." The NRC inspectors verified that each of those procedures were updated with the new requirements.

<u>(Closed) Notice of Violation VIO 07200041/2018-001-04, Failure to provide adequate instructions in procedures, 10 CFR 72.150, EA-18-155</u>

The NRC Special Inspection Report documented a violation of NRC requirements related to the licensee's failure to prescribe activities affecting quality by documented instructions or procedures of a type appropriate to the circumstances and include appropriate quantitative or qualitative acceptance criteria for determining that important activities had been satisfactorily accomplished.

The licensee responded to the Notice of Violation and described the corrective steps taken to ensure full compliance in SCE submittal to the NRC, dated December 26, 2018 (ADAMS Accession ML18362A148). The inspectors reviewed the licensee's implemented corrective actions related to procedural direction during follow-up inspection activities. The inspectors concluded, based on the changes described above, that the licensee had performed adequate corrective actions to restore compliance, address extent of condition, and prevent recurrence.

However, the inspectors made observations related to the corrective actions to improve Holtec Procedure HPP-2464-400 (see Section 1.2.3.c.(2)). The licensee subsequently addressed the NRC observations. No additional deficiencies were identified during NRC's review of this violation.

This closes VIO 07200041/2018-001-04, "Failure to provide adequate instructions in procedures" (10 CFR 72.150), EA-18-155.

(3) <u>Training</u>

Inadequate training was identified by the licensee as a contributing cause that led to the canister misalignment event. Specifically, RCE QI-2529 Contributing Cause 5

stated, in part, that the "Holtec training program did not consider uniqueness of UMAX system relative to HI-STORM or uniqueness of challenges raised in a long-term project which led to not fully establishing qualification or proficiency requirements for the Task Performers when transferring a canister into a UMAX system."

The licensee had several corrective actions associated with training, for both fuel handling personnel and oversight personnel, which broadly included: updated initial training, on-the-job demonstrations, updated qualifications, ongoing proficiency requirements, updated training lesson plans, scripted pre-job briefs, and the incorporation of site-specific operating experience into the training program. The specific corrective actions associated with training included:

- CA-19 and CA-20: Developed a SONGS site-specific training program and procedures which augmented the existing Holtec corporate training program and procedures. The corrective actions required that the site training program to include a site-specific task list and a task to training matrix which described all the applicable positions of a fuel handling crew to be utilized at SONGS. The corrective actions required all positions to be described and minimum training and qualifications for each position listed. The training program was required to include the appropriate elements of a systematic approach to training (SAT).

- CA-22: Included a 10 CFR 72.48 evaluation to incorporate additional text into Chapter 9 of the FSAR to add criteria for load limits, training, procedure compliance, and use of engineering features.

- CA-23: Required the addition of a training consultant to perform an evaluation of the current site-specific training program, including effectiveness, and to provide recommendations for improvements to the Holtec standard training program. Areas of evaluation included, but were not limited to, review and enhancement of task analysis matrices, the development of training programs, implementation plans, proficiency requirements, and requalification requirements.

- CA-24: Required training and qualification for all loading personnel currently assigned to the project in accordance with new SONGS site-specific training program requirements (CA-20).

The licensee concluded that procedure HSP-34, "Training of Subcontracted Field Service Personnel," which was previously used to train and qualify the pool-to-pad personnel, was not based on a SAT. A site-specific training program, HPP-2464-1134, "Training of site services personnel," Revision 1, was developed by the licensee and reviewed by the inspectors. This SAT based program was developed to be used in conjunction with procedure HSP-34.

A SAT program is defined in 10 CFR 55.4, and includes the following attributes: (1) systematic analysis of job performance requirements and training needs; (2) the derivation of learning objectives, based upon the preceding analysis, which describe desired performance after training; (3) the training program design and implementation based on the learning objectives; (4) the evaluation of trainee mastery of learning objectives during training; and (5) the training program evaluation and revision based upon the performance of trained personnel in the job setting.

The new site-specific training procedure HPP-2464-1164 required:

- All positions to be described and minimum training and qualifications for each position listed in a training matrix.
- To contain the minimum qualification requirements to ensure that personnel were appropriately trained prior to performing fuel transfer activities.
- To include the appropriate elements of a SAT program.

The training corrective actions required the licensee to update all lesson plans, which included an additional 13 new lesson plans and development of seven new on-the-job training requirements using the SAT process. The corrective action program and Operating Experience (OE) programs were included as a feedback loop into the training program as required by procedure HPP-2464-1164. In addition, the licensee staffed a site program training manager to oversee the training program and ensure the SAT program elements were maintained. Finally, the inspectors reviewed the changes in UMAX FSAR, Chapter 9, to verify the change included revised language from CA-22.

(Closed) Notice of Violation VIO 07200041/2018-001-03, Failure to assure that operations of important-to-safety equipment were limited to trained and certified personnel, 10 CFR 72.190, EA-18-155

The NRC Special Inspection documented a violation of NRC requirements related to the licensee's failure to assure that operation of equipment and controls, that had been identified as ITS in the Safety Analysis Report, were limited to trained and certified personnel or were under the direct supervision of an individual with training and certification in the operation.

The licensee submitted a response to the NRC on December 26, 2018 (ADAMS Accession ML18362A148), which contained the corrective steps taken to ensure full compliance was achieved. The inspectors reviewed the licensee's implemented corrective actions related to the training of personnel during follow-up inspection activities. The inspectors concluded, based on the changes described above, that the licensee had performed adequate corrective actions to restore compliance, address the extent of condition, and prevent recurrence. No additional deficiencies were identified during the inspectors' review of this violation.

This closes VIO 072-00041/2018-001-03, "Failure to assure that operations of important-to-safety equipment were limited to trained and certified personnel" (10 CFR 72.190), EA-18-155.

(4) Equipment and Personnel

The licensee's causal evaluation contained corrective actions to implement a new load monitoring system, increased the number personnel present during downloading operations, and added remote monitoring capabilities to limit canister misalignments and prevent a condition in which the lifting devices no longer controlled the weight of the canister.

The new load monitoring equipment included two load sensing shackles, which were placed in-line with each respective downloading sling. These dual and redundant load sensing shackles were calibrated by an approved vendor to an accuracy of ±1% of the actual weight. The load sensing shackles wirelessly transmitted the weight of the canister to two digital readout tablets. Each tablet was equipped with an audible and visual alarm that would activate when the weight decreased below the established set points. One tablet was positioned next to the Holtec cask loading supervisor and SCE oversight specialist. The second tablet was positioned above the VCT control box and could be observed by both the VCT operator and an additional spotter, who was required to be on the VCT platform during downloading operations.

As part of the equipment enhancements, the licensee installed a camera on the side of one of the VCT towers. The camera was positioned to provide an overhead view of the top of the canister as it passed through the transfer cask into the ISFSI vault. The camera wirelessly displayed the video feed to a monitor that was located next to the Holtec cask loading supervisor and the SCE oversight specialist.

Other enhancements included increased number of personnel on the ISFSI pad during downloading operations from the two personnel (VCT operator and rigger in the man-basket) during the August 3rd incident to nine individuals on the ISFSI pad. This included an additional rigger in a separate elevated lift-basket to visually observe the canister as it was lowered through the transfer cask into the ISFSI vault.

During the downloading demonstrations performed by the licensee January 28 through February 1, 2019, the NRC inspectors observed the licensee successfully utilize the new equipment to safely lower a canister into the ISFSI vault. However, the inspectors identified two violations of NRC requirements regarding the licensee's equipment implementation and procurement of the new load monitoring equipment (see Section 1.2.3.c.(3) and (4)).

(5) Corrective Action Program

The licensee's ACE 0818-20356, RCE QI-2529, and CCE 0618-77146 identified corrective actions to address deficiencies in the CAP. The ACE 0818-20356 identified that ISFSI project management had not encouraged initiation of condition reports for deviations experienced in dry cask storage operations as a contributing cause. The RCE QI-2529 identified that Holtec had not fostered an environment that promoted sharing of internal and external operating experiences among the dry cask storage workers. The CCE 0618-77146 identified Holtec procedures and processes that input to the Holtec Field Condition Report (FCR) process and the Holtec CAP, were not sufficiently detailed or prescriptive to guide or instruct a person with limited quality assurance (QA) and CAP experience to identify, and effectively resolve, conditions adverse to quality and/or trends in a timely manner.

To address these issues, all three of these causal evaluations recommended corrective actions in the area of the CAP which included the following actions: (1) conducting a lessons learned case study based on recent events to clarify condition report initiation; (2) developing oversight specialist condition report training; (3) revising procedure HSP-42, "Project Manager's Desktop guide for Site Services Pool to Pad Projects," to include a section on operational experience; (4) revising procedure HSP-35, "Procedure for Field Condition Reports and Procedure Field

Change Notices for All Site Work," to provide clarification on the threshold for condition report initiation; (5) establishing a process to ensure operational experiences were communicated across and within project areas; (6) assigning a qualified and experienced full time Holtec QA Manger to the ISFSI Project to oversee the CAP; (7) developing a SCE CAP training plan; and (8) requiring Holtec to adopt and adhere to SCE's CAP for SONGS related work activities.

Action (1) required SCE to develop a personnel training module that included specific events identified during active fuel transfer operations that provided lessons learned applicable to improving SCE's implementation of its CAP. The training developed by SCE included examples of deviations experienced during the loading campaign and at other sites as well as the August 3, 2018, downloading operations. The inspectors reviewed the training documentation and verified that applicable dry cask storage staff had completed the required training.

Action (2) involved training the SCE oversight specialists in documenting issues into the oversight specialist database. The training emphasized the documentation of relevant issues or comments into the database with sufficient detail such that the observed deficiencies could be understood. The inspectors reviewed the training documentation and verified the roster of ISFSI oversight specialists had completed the required training.

Action (3) revised procedure HSP-42 to include steps which required operating experience, lessons learned, and best practices encountered during the execution phases of fuel loading operations to be captured by the Holtec project manager. Six sources of operating experience were identified: (1) standard shift turnover sheets; (2) FCRs; (3) management observation program comments; (4) site services weekly project updates/conference calls; (5) the Holtec Users Group database; and (6) the Holtec Lessons Learned database. The operating experience collected from these sources was required to be shared with dry cask storage workers during pre-job briefings and two-minute drills, as applicable, by the Holtec site project manager.

Action (4) revised procedure HSP-35 to provide procedural clarification on the threshold for initiating an FCR. The definitions section of procedure HSP-35 was expanded to include "Short-term Operations." A procedure step was included that explained that "any observed event during Short-term Operations that indicated an abnormal or unexpected condition shall be entered into the FCR tool for further evaluation."

Action (5) revised procedure HSP-42 to require the project manager to collect and disseminate pertinent operating experience to the appropriate dry cask storage personnel on a routine basis. This corrective action also relied on changes made to procedure HSP-35, which lowered the threshold for FCR reporting; SCE CAP training, which redefined the lower thresholds for problem identification; and procedure HSP-1101, "Procedure for Project Risk Management," which was revised to include lessons learned and operating experience documentation that must be reviewed for potential risk impacts.

Action (6) appointed a QA manager for Holtec to the SONGS facility. The appointee had experience with 10 CFR Part 50, Appendix B, and 10 CFR Part 72, Subpart G, requirements. The quality manager tasks included actions to improve quality in work

performed at SONGS, interface with Holtec personnel, maintaining high standards for Holtec work activities, performing corrective action evaluations, performing trending on FCRs, and addressing quality related issues as they are identified on site. The NRC inspectors reviewed the new quality assurance manager's resume and confirmed the individual had the knowledge and experience to perform the required responsibilities.

Action (7) required CAP training to be provided to site personnel. The NRC reviewed lesson plans and attendance records. The training lesson plans contained all the required information described in the causal evaluation and included additional enhancements to strengthen the CAP.

Action (8) required all workers, including contractors, to use the SCE CAP for activities on site. The NRC reviewed the revised process, which included an organization chart to identify which onsite personnel would have access to SCE's Action Request system and documentation that showed Holtec managers and workers had been provided credentials to access the Action Request system.

(Closed) Notice of Violation VIO 07200041/2018-001-01, Failure to identify and correct conditions adverse to quality (10 CFR 72.172), EA-18-155

The NRC Special Inspection documented a violation of NRC requirements related to the licensee's failure to establish measures to ensure that conditions adverse to quality, such as failures, malfunctions, deficiencies, and deviations were promptly identified and corrected.

The licensee submitted a response to the NRC on December 26, 2018 (ADAMS Accession ML18362A148) which contained the corrective actions taken to ensure full compliance was achieved. The inspectors reviewed the corrective actions implemented related to the use of the licensee's corrective action program during follow-up inspection activities. The inspectors concluded, based on the changes described above, that the licensee had performed adequate corrective actions to restore compliance, address the extent of condition, and prevent recurrence. No additional deficiencies were identified during the inspectors' review of this violation.

This closes VIO 07200041/2018-00101, "Failure to identify and correct conditions adverse to quality" (10 CFR 72.172), EA-18-155.

(6) Reportability

The licensee performed a reportability root cause evaluation (RRCE 1218-33805) to evaluate their failure to make an event notification to the NRC Operations Center for the August 3, 2018, misalignment incident. The corrective actions to address the identified causes included the following actions: (1) developing 10 CFR 72.75 training that identified ITS components, potential accidents, and failures that influence reportability; (2) establishing requirements for biennial refresher training; (3) conducting reviews to determine potential reportability requirements related to other site activities; (4) conducting reviews to determine the target audience for training the reportability changes; (5) revising site notification procedures to have a more conservative reporting bias and the identification of the Shift Manager as the individual responsible for the final decision on reportability for the site; (6) developing and conducting a case study with licensee managers and regulatory assurance

personnel on the communications and reportability aspects of the August 3, 2018, incident; and (7) conducting all-hands briefings regarding the reportability violation and future expectations for reporting.

For actions (1) through (4), SCE developed 10 CFR 72.75 training and required biennial refresher training. This training was delivered to SCE managers and Regulatory Assurance personnel. The training included discussions of accidents and design basis events for both the UMAX and NUHOMS ISFSI designs. The training included the descriptions and function of ITS structures, systems, and components and potential failures that would require reporting under 10 CFR 72.75. The training and biennial refresher requirements were included under the Shift Manager/Certified Fuel Handler Training Program. The initial target audience was SCE managers and Regulatory Assurance staff.

Action (5) required that SCE revise procedure SO-123-0-A7, "Notification and Reporting of Significant Events," to have a conservative bias toward reporting requirements. The procedure was revised to include guidance that if the condition being considered did not literally meet the reporting criteria, but was close, then the staff was directed to make a voluntary report using the closest reporting requirement that matched the condition under consideration. This was required to be completed within the time-frame stipulated by the reporting requirement. Procedure SO-123-0-A7 was also revised to encourage the voluntary reporting of any event or condition that could have safety significance or represent a generic concern.

The reporting procedure was further revised to identify the Shift Manager as the site individual responsible for making the final decision on reportability. Lastly, the SCE notification procedure was revised to include Attachment 11, "Reportability Determination," for a decision-making flow-chart. The flow-chart required the Shift Manager to chair a Reportability Management meeting/conference call to discuss potential reporting conditions. The call decision was required to be documented with the date and time of the decision, the start-time of the reportability clock, when the report was due, and the date/time the event notification was made.

Action (6) required the licensee to develop a case study training module that covered the specifics of the August 3, 2018, misalignment incident and the contributing factors that led to the licensee's failure to properly assess the event and to report the incident to the NRC Operations Center, as required by 10 CFR 72.75(d)(1). The case study discussed the specific details of the incident, acknowledged missed opportunities, and provided examples of how the notification procedure was revised to prevent recurrence of the notification failure. The case study required attendees to fill out a work-sheet that asked specific questions related to the event.

Action (7) required that the Chief Nuclear Officer provide an all-hands briefing to SCE staff and a separate briefing to SCE managers to discuss the violation. The briefings were to discuss the licensee's failure to make the 24-hour NRC notification, the causes of the failure, and management expectations for a conservative bias when making reportability decisions moving forward.

(Closed) Notice of Violation VIO 072-00041/2018-001-05, "Failure to make 24-hour notification" (10 CFR 72.75), EA-18-155

The NRC Special Inspection documented a violation of NRC requirements related to the licensee's failure to make a required 24-hour notification to the NRC within the required timeframe.

On November 8, 2018, the licensee issued Licensee Event Report (LER) 2018-001-0 (ADAMS Accession ML18317A060) in accordance with 10 CFR 72.75(d)(1) for the event and restored compliance. The licensee submitted its response to the Notice of Violation, on April 23, 2019 ADAMS Accession ML19116A056), which contained the corrective actions taken to ensure full compliance was achieved.

The NRC concluded that SCE's completed and proposed corrective actions, as described above, restored compliance, addressed extent of condition, and were adequate to prevent recurrence. No additional deficiencies were identified during the inspectors' review of this violation.

This closes VIO 072-00041/2018-001-05, "Failure to make 24-hour notification," (10 CFR 72.75), EA-18-155.

c. Observations and Findings

(1) Executive Oversight Board

The inspectors observed that CAPR-1 associated with the RCE QI-2529 appeared to be administrative in nature and did not meet the level of rigor associated with a CAPR, which should serve to preclude repetition of significant performance issues. The CAPR assigned changes to the Executive Oversight Board agenda to provide an increased focus on early identification of challenges to the project to ensure issues were properly resolved before undesired events occurred.

In response to the inspectors' observation, the licensee placed the identified observation into the corrective action program as AR-0818-7655. The licensee bolstered the required changes to the Executive Oversight Board agenda to incorporate additional techniques to review Management Review Meeting data, participation to evaluate current performance against risk registers, evaluate industrial safety trends, review quality metrics, and review SCE oversight effectiveness. The changes provided rigor to the agenda which served to consistently evaluate project performance against pre-determined standards. The NRC inspectors reviewed the new meeting agenda to verify the topics reviewed would ensure early identification of challenges to the project. Based on the licensee's changes and level of detail that would be reviewed during the meetings, the NRC concluded that the changes were appropriate to support early identification of significant performance deficiencies.

(2) Downloading Procedure

The inspectors determined that SCE had made substantial improvements to fuel handling procedures to ensure safe operations. However, the NRC identified that notable procedural weaknesses remained in downloading procedure HPP-2464-400 "MPC Transfer at SONGS," Revision 17. Procedure weakness included: (1) missing contingency steps for potential new equipment failures; (2) while there were some

criteria specified for when to suspend downloading operations, not all scenarios were addressed; and (3) the procedure lacked some steps necessary to maintain seismic qualifications during cask transport from the fuel building to the spent fuel storage pad.

In response, the licensee initiated AR 0119-81239-10 and AR 0119-81239-9 to capture the inspectors' observations. The licensee took corrective actions and addressed the identified omissions in the next procedural revision.

(3) Equipment Designation

Corrective action CA-1, associated with ACE 0818-20356, implemented guidance for a load monitoring device to ensure load indication was available to assist with suspending operations if the load was lost. SCE implemented the design change to incorporate the new load monitoring equipment using Nuclear Engineering Change Package (NECP) 0918-64884, "VCT Live Load Monitoring System," Revision 1. The load monitoring equipment included intermediate slings, a master link, and load sensing shackles which would be placed in-line with each of the ITS downloading slings. The inspectors identified that the NECP inappropriately designated the new load monitoring equipment as not-important-to-safety (NITS). Inspectors determined that since the new equipment was to be placed in-line with existing ITS downloading equipment, the new equipment, which failure could result in the drop of a loaded canister, should be controlled and designated under SCE Quality Assurance Program as ITS equipment.

10 CFR 72.146(a) states, in part, the licensee shall establish measures to ensure that the design bases are correctly translated into specifications, drawings, procedures, and instructions. These measures must include provisions to ensure that appropriate quality standards are specified and included in design documents.

Contrary to the above, on December 7, 2018, the licensee failed to establish measures to ensure that the appropriate quality standards were specified and included in design documents. Specifically, the licensee inappropriately designated the new load monitoring equipment at the wrong quality standard in NECP 0918-64884-1, Revision 1.

This violation was dispositioned per the traditional enforcement process using Section 2.3 of the NRC's Enforcement Policy. The NRC determined that the finding was of low safety significance since the equipment had not been used with any loaded canisters and the load monitoring equipment had been purchased by the vendor at the appropriate quality assurance designation of ITS. This finding was determined to be of more than minor safety significance since if left uncorrected, the deficiency could lead to a more significant safety concern.

Consistent with the guidance in Section 1.2.6.D of the NRC Enforcement Manual, if a violation does not fit an example in the Enforcement Policy Violation Examples, it should be assigned a severity level: (1) commensurate with its safety significance; and (2) informed by similar violations addressed in the Violation Examples. The violation was evaluated to be similar to Enforcement Policy Section 6.5.d.2.

The licensee entered the issue into the CAP as AR 1218-20333. The licensee restored compliance by verifying that the load monitoring equipment met all applicable

industry standards of NUREG 0612 and American National Standards Institute (ANSI) N14.6 requirements to meet the ITS qualification and revised the design change package to include the correct designation. Additional corrective actions taken by the licensee to preclude repetition included: performing an event investigation, conducting training for the SCE engineering team, conducting reviews of implementing procedures, and updating the site's Quality Equipment List. Because the licensee entered the finding into the CAP, the safety significance of the issue was low, and the issue was not repetitive or willful, this Severity Level IV violation was treated as a non-cited violation (NCV), consistent with Section 2.3.2.a of the Enforcement Policy (NCV 07200044/2018-002-01, Failure to ensure appropriate quality standards (10 CFR 72.146)).

(4) Equipment Procurement

The NRC inspectors reviewed all the procurement documents associated with the new load monitoring equipment that was described in NECP 0918-64884-1. This included reviewing the Holtec purchase specifications and equipment's certificate of conformance for each of the new components (load sensing shackles, master links, and intermediate slings).

The weight of the loaded canister, rigging equipment, and an additional 15% dynamic factor was calculated to be 118,640 lbs (59.34 tons) per HI-2156458 "Cask Handling Weights at SONGS," Revision 1. Each side of the rigging was required to be able to handle the load in the event that one side fails. This would require all rigging on each side to have a minimum rating of 59.34 tons.

The inspectors identified an issue with the certificate of conformance for the StraightPoint load sensing shackles. The load sensing shackles were rated to the capacity of 185,000 (92.5) tons, which was well above the required rating. However, the Holtec Purchase Specification PS-223 "Procurement Specification for Significant Rigging," Revision 0, Step 7.0, "Special Tests," required a proof test load of twice the rated vertical capacity to all rigging components. This is also required by common industry rigging standards contained in American Society of Mechanical Engineers (AMSE) B30.26 "Rigging Hardware," Section 1.4.2. The inspectors identified that the load sensing shackles were only load tested to 1.5 times the rated capacity instead of the required twice the rated capacity per purchase specification PS-223.

Additionally, Holtec's Approved Vendor List, contained the following restriction, "lifting equipment load testing must be performed at Aston I&I Sling factory." The inspectors observed that the proof load testing for the new load sensing shackles was performed at the manufacturer's facility (StraightPoint) and not by Aston I&I Slings factory per Holtec's Approved Vendor List's restrictions.

10 CFR 72.154(a) states, in part, the licensee shall establish measures to ensure that purchased material, equipment, and services, whether purchased directly or through contractors and subcontractors, conform to the procurement documents.

Contrary to the above, on December 7, 2018, the licensee failed to establish measures to ensure that purchased equipment conformed to the procurement documents. Specifically, the licensee accepted the StraightPoint load sensing shackles, which had not been proof load tested to twice the rated capacity as specified in Holtec Purchase

Specification PS-223, Step 7.0. Additionally, the licensee failed to ensure the proof load testing was performed by an approved vendor.

This violation was dispositioned per the traditional enforcement process using Section 2.3 of the NRC's Enforcement Policy. The NRC determined that the finding was of low safety significance since the equipment had not been used with any loaded canisters. This finding was determined by inspectors to be of more than minor safety significance because, if left uncorrected, the deficiency could lead to a more significant safety concern.

Consistent with the guidance in Section 1.2.6.D of the NRC Enforcement Manual, if a violation does not fit an example in the Enforcement Policy Violation Examples, it should be assigned a severity level: (1) commensurate with its safety significance; and (2) informed by similar violations addressed in the Violation Examples. The violation was evaluated to be similar to Enforcement Policy Section 6.5.d.2.

The licensee entered the issue into the CAP as AR 1219-52380. The licensee restored compliance by having the load sensing shackles proof tested to twice the rated capacity in accordance with purchase specification PS-223, by the Aston I&I Slings factory per Holtec's Approved Vendor List's restrictions. Additional corrective actions taken by the licensee to preclude repetition included: performing an apparent cause evaluation, reviewing other procured equipment documentation from Aston I&I Slings to ensure testing requirements were met, developing a revised SONGS rigging program to require an independent review and approval of vendor ITS rigging documentation, creating a project specific purchase specification for downloading shackles to provide clear details on load testing requirements, and conducting training for SCE site service project managers. Because the licensee entered the finding into the CAP, the safety significance of the issue was low, and the issue was not repetitive or willful, this Severity Level IV violation was treated as a NCV, consistent with Section 2.3.2.a of the Enforcement Policy (NCV 07200044/2018-002-02, Failure to ensure purchased material conformed to the procurement documents (10 CFR 72.154)).

d. Conclusions

Based on the licensee's evaluations and actions taken in the areas of licensee oversight, procedures, training, equipment, corrective action program, and reportability, the inspectors concluded that the corrective actions implemented were appropriate to prevent recurrence of the issues and were adequately prioritized with consideration of the risk significance and regulatory compliance. The inspectors concluded that SCE's completed corrective actions in the areas of training, corrective action program, and procedures restored compliance for the violations document in the NRC Notice of Violation issued in the NRC Special Inspection Report.

Additionally, the licensee's corrective actions taken to address the violation for failure to make a report to the NRC, documented in NRC letter of Notice of Violation and Civil Penalty, were adequate to restore compliance and prevent recurrence. However, during the NRC's review of the corrective actions taken, the inspectors identified two observations and two violations of NRC requirements related to the licensee's corrective actions. The licensee took adequate corrective action to restore compliance on the issues identified through the CAP. The violations were determined to have low safety

significance and the Severity Level IV violations were treated as NCVs. As a result, Inspection Objective 3 was met.

1.2.4. Corrective Actions Planned (Objective 4)

a. Overview

The inspectors evaluated whether the corrective actions planned to address and preclude repetition of significant performance issues were prompt and effective, and that appropriate quantitative or qualitative measures of success had been developed for determining the effectiveness of planned corrective actions.

b. Assessment

The licensee's causal evaluations contained effectiveness assessments to validate that the corrective actions were successful. In the area of training, the licensee's corrective action plan included acquiring a training consultant to perform an evaluation of the new site-specific training program, including effectiveness, and develop recommendations for improvement. The recommendations would support training enhancements for the SONGS training program and the vendor's standard training program. The area to be evaluated included task analysis matrices, training program, implementation plans, proficiency requirements, and requalification requirements.

In the area of operations, an effectiveness review schedule was established to assess the effectiveness of all corrective actions during both dry run demonstrations/training evolutions and during actual fuel movement activities. The review included an assessment of trends in lifting activities, verification of trained personnel, and detailed observational surveillance of lifting activities by independent auditors. The surveillance tasks included a review of training verification, procedure proficiency, adequate use of the CAP, and verification of management observations.

The licensee's oversight effectiveness review included corrective actions to conduct additional procedure reviews to identify new technical deficiencies, review of oversight task guides to verify sufficient guidance and enhancements, and various peer observations of oversight individuals to verify proficiency in procedures, task guide knowledge, initiation of corrective actions, and ensure desired behaviors. The effectiveness review actions contained detailed criteria that an independent assessor was required to verify during the dry-run exercises and during continued fuel loading activities.

In the area of reportability, the licensee's corrective actions included a new real time reporting exercise to be conducted monthly. All applicable individuals would be required to participate in the exercise. The exercises would take place for three consecutive months and success would be based on no incorrect reportability determinations. In addition, the new reportability process required the assignment of a "meeting skeptic" to monitor the reportability meetings to ensure the desired behavior changes continued and adequate determinations were made.

c. Observations and Findings

No findings were identified with the licensee's corrective actions planned.

d. Conclusion

Based on the licensee's evaluations and documented actions planned, the inspectors concluded that the licensee had adequately established measures to validate the effectiveness of the corrective actions to prevent recurrence. As a result, Inspection Objective 4 was met.

2 Follow-up of Events and Notices of Enforcement Discretion (IP 71153)

2.1 Inspection Scope

The inspectors evaluated licensee events to verify the licensee's corrective actions were adequate to restore compliance. The inspectors reviewed LERs to ensure the reports were timely, accurate, and the required corrective actions had been completed. Additionally, inspectors documented review of follow-up items from the NRC Special Inspection Report.

2.2 Assessment

2.2.1 (Closed) Licensee Event Report 2018-001-0, Spent Nuclear Fuel Canister Temporarily Wedged in Dry Cask Storage Container

On November 8, 2018, the licensee issued LER 2018-001-0 (ADAMS Accession ML18317A060) in accordance with 10 CFR 72.75(d)(1) and (g) for inadvertently disabling redundant ITS slings while lowering a spent fuel canister into the ISFSI on August 3, 2018.

The NRC Special Inspection Report, dated December 19, 2018, documented three cited violations and two apparent violations associated with this event that were handled through the NRC's escalated enforcement process.

During this supplemental inspection, the NRC inspectors reviewed the planned and implemented corrective actions taken by the licensee for the identified violations and determined the actions to be adequate to restore compliance and prevent recurrence.

This LER is closed.

2.2.2 (Discussed) Licensee Event Report 2018-002-0, Spent Nuclear Fuel Transport Conveyance Vehicle Operated Outside Obstacle Clearance Limits

On February 14, 2019, the licensee issued LER 2018-002-0 (ADAMS Accession ML19050A170) in accordance with 10 CFR 72.75(d)(1) and (g) for past operations of the low-profile-transporter. The licensee identified that transporter's center of gravity was not maintained within limitations specified in the site's specific analysis and operations had been conducted too close to adjacent structures (light posts) and was outside the calculated clearance limits specified in the site's seismic analysis. The licensee identified that the site procedures did not provide sufficient detail to comply with the seismic stability calculation. No actual incidents with structures or collisions with obstacles occurred during past fuel transfer operations and there was no impact to plant personnel or public health and safety.

As part of the licensee's extent of condition review associated with licensee causal evaluation RRCE 1218-33805, the licensee notified the NRC Operations Center within 24 hours of discovery of the issue (Event Notification 53798) and submitted an LER to the NRC within the 60-day time limit in accordance with 10 CFR 72.75(d)(1) requirements.

As part of the review of the August 3, 2018, event, the inspectors reviewed the licensee's corrective actions to restore compliance and prevent recurrence. This included reviewing the licensee's updated seismic analysis which determined that the variance in the height of the conveyance, during the past operations was acceptable and the licensee's changes made to the transportation procedures. Additionally, the inspectors observed licensee perform dry run exercises that demonstrated the procedural changes were adequate to ensure the conveyance would remained within the bounds and limitations of the analysis (see Section 2.2.8). However, as reported in the LER, the licensee was still in progress of developing an analysis to determine if the operation of the conveyance with the reduced obstacle clearance was acceptable. Thus, this LER will remain open, pending NRC review of this additional information.

2.2.3 (Closed) NRC Event Notification #53858, Inadequate Analysis for VCT Operations

During the on-site portion of this inspection, the NRC inspectors observed demonstrations of the licensee's corrective actions associated with downloading operations. As the VCT approached the mating device, the procedural steps directed the removal of the restraint band from around the HI-TRAC VW transfer cask. As operations continued, the transfer cask was raised and continued to travel approximately 15-20 feet before being lowered onto the mating device to allow downloading operations to begin. While traveling without the restraint band, the transfer cask was visibly rocking as the VCT approached the mating device. The inspectors questioned the licensee during the site observations to determine if the site's seismic analysis addressed and evaluated travel of the loaded HI-TRAC VW without the restraint band.

On February 2, 2019, in accordance with 10 CFR 72.75(d)(1) the licensee notified the NRC Operations Center within 24 hours of the discovery of issues regarding the past use of the VCT to transport spent fuel storage canisters to the ISFSI pad. The licensee reported that over short periods of time, the canister transport process utilizing the VCT could have been operated without a supporting seismic analysis while transporting loaded canisters for storage. The licensee subsequently retracted Event Notification #53858 on April 2, 2019, citing a revised seismic calculation which confirmed the transport process and VCT operations met the seismic requirements of the Holtec Certificate of Compliance.

The licensee's failure to follow the initial site specific seismic analysis was determined by inspectors to be a violation of NRC requirements. This event notification is closed (see Section 2.2.4 below).

2.2.4 Finding related to the Licensee's Event Notification

The licensee's event notification EN #53858 documented that past VCT operations had not been conducted within the requirements of seismic evaluation HI-2156626, "VCT Stability Analysis on Route to ISFSI Pad and on ISFSI Pad for SONGS," Revision 3. For short periods of time, the VCT seismic restraint band was prematurely removed from the

transfer cask prior to stack-up evolutions. Evaluation HI-2156626, Section 4.0, "Assumptions," stated that, "the transfer cask and the VCT were considered to behave as a rigid body." The evaluation conservatively assumed the seismic restraint band, which braced the transfer cask to the VCT, was in position at all times during transportation operations.

10 CFR 72.212(b)(3), requires, in part, that the general licensee shall ensure that each cask used conforms to the terms, conditions, and specifications of a Certificate of Compliance as listed in 10 CFR 72.214.

10 CFR 72.214 states, in part, that Certificate Number 1040 [Docket Number 072-01040] Amendment Number 2, effective date January 9, 2017, is an approved cask for storage of spent fuel under the conditions specified in the Certificate of Compliance for the Holtec HI-STORM UMAX Storage System.

Certificate of Compliance 072-01040, Appendix B Technical Specification 3.4.15 requires, in part, the loaded transfer cask and its conveyance shall be evaluated to ensure, under the site-specific Design Basis Earthquake (DBE), that the cask and its conveyance does not tip-over or slide off the haul route.

Contrary to the above, from January 30, 2018, to August 3, 2018, the licensee failed to ensure the cask and its conveyance was evaluated under the site-specific DBE. Specifically, the NRC identified that past VCT transportation operations were not evaluated under the site-specific DBE, since operations were conducted outside the requirements in seismic evaluation HI-2156626.

This violation was dispositioned per the traditional enforcement process using Section 2.3 of the NRC's Enforcement Policy. The NRC determined that the finding was of low safety significance since the licensee had re-performed the evaluation, addressed the deviation that occurred, and demonstrated the canister and its conveyance would not have tipped over or slipped off the haul route during those transportation operations due to prematurely removing the seismic restraint band. This finding was determined by inspectors to be of more than minor safety significance, since if left uncorrected, the deficiency could lead to a more significant safety concern.

Consistent with the guidance in Section 1.2.6.D of the NRC Enforcement Manual, if a violation does not fit an example in the Enforcement Policy Violation Examples, it should be assigned a severity level: (1) commensurate with its safety significance; and (2) informed by similar violations addressed in the Violation Examples. The violation was evaluated to be similar to Enforcement Policy Section 6.1.d.1.

The licensee entered the finding into the CAP as AR 0219-88442, 0219-22465, and 0319-95843. The licensee restored compliance by revising the site-specific seismic analyses to bound transportation operations conducted at the site. Additional corrective actions taken by the licensee to preclude repetition included: performance of an apparent cause evaluation, submittal of formal reports to the NRC in accordance with 10 CFR 72.75(d)(1), conducted training on the lessons learned, briefed the Holtec Users Group, and revised the process used to transmit vendor information to the NRC to require a documented review by the appropriate SONGS organization prior to transmittal. Because the licensee entered the issue into the CAP, the safety significance of the issue was low, and the issue was not repetitive or willful, this Severity Level IV

violation was treated as a NCV, consistent with Section 2.3.2.a of the Enforcement Policy (NCV 07200044/2018-002-03, Failure to ensure the loaded transfer cask and its conveyance was evaluated under the site-specific DBE (10 CFR 72.212)).

2.2.5 Follow-up of Special Inspection Charter Items from the NRC Special Inspection

a. Drop Evaluation

The inspectors independently reviewed licensee's evaluation to analyze the potential effects of a canister drop. The licensee evaluation was documented in evaluation HI-2188261 "Structural Evaluation of the MPC Handling Event at SONGS," Revision 3. Evaluation HI-2188261 conservatively assumed the canister fell, uninterrupted, 25 feet to the base of the UMAX vault. The actual height the canister potentially could have dropped was 18 feet. The evaluation defined a canister breach as the point at which the strain measured at any location exceeded the specified strain limit for the material.

Following the guidance from NUREG-1864 "A Pilot Probabilistic Risk Assessment of a Dry Cask Storage System at a Nuclear Power Plant," dated March 2007, the evaluation considered the effects of strain rate and temperature, using a strain in the weld material to be estimated at 0.73 in/in (extension length/original length). Conservatively, the evaluation used one standard deviation below the allowable strain to establish a limit of 0.55 in/in for the weld material. The 316 stainless steel base material had an even higher acceptable strain limit. Conservatively, the evaluation limited the strain of the base material to 0.55 in/in as well.

The drop analysis was performed using the finite element code LS-DYNA, which has been validated under Holtec's Quality Assurance Program, and was a method of evaluation that had been used in the UMAX FSAR for other canister analyses. The results of the analysis resulted in a maximum computed effective strain of 0.468 in/in, which was below the conservative limit of 0.55 in/in for both the base metal and weld material. NRC inspectors independently reviewed the analysis and concluded that the canister would not have breached had the canister fallen 18 feet to the bottom of the UMAX vault.

The condition of the fuel after the postulated drop and the canister's ability to continue to perform its safety function in the regards of pressure, thermal, criticality control, and shielding was analyzed in evaluation HI-2188261, and Storage Position Paper DS-470, "Expected Fuel Damage after MPC Drop," dated November 6, 2018. The analysis concluded that the damage would be mostly limited to deformation and buckling of lowest section of the fuel rods of the spent fuel assemblies. The inspectors independently reviewed each safety function analysis for accident conditions with regard to criticality, thermal performance, shielding, and pressure.

The inspectors concluded that expected temperature and pressure limits would have remained under the accident limits described in FSAR, criticality safety would have been maintained since the confinement boundary was not breached and the system remained dry, and external radiological dose rates of the canister, located in the vault, would have minimal increases. However, the condition of fuel after the postulated drop would not meet the licensing requirements for storage or transportation. The licensee would be required to perform either significant evaluations or supplemental operations to ensure

the safe retrieval, unloading, and re-packaging of the fuel while minimizing the dose to personnel.

b. Scratch Evaluation

As part of the corrective actions from the ACE 0818-20356 and RCE QI-2529, actions were taken to address the discrepancies within the UMAX FSAR, specifically the incidental contact that occurs when a canister was downloaded into the UMAX vault. The UMAX FSAR, Revision 4, Sections 1.2.4 and 9.5 vii, contained design statements that stated:

- Section 1.2.4, "Operational Characteristics of HI-STORM UMAX," *The vertical insertion (or withdrawal) of the MPC eliminates the risk of gouging or binding of the MPC with the CEC parts*

- Section 9.5 vii, "Regulatory Compliance," *Because the MPC insertion (and withdrawal) occurs in the vertical configuration with ample lateral clearances, there is no risk of scratching or gouging of the MPC's external surface (Confinement Boundary). Thus, the ASME Section III Class 1 prohibition against damage to the pressure retaining boundary is maintained.*

The HI-STORM UMAX MPC-37 used at SONGS is made of a type 316 stainless steel. It is approximately 76 inches in diameter and 17 feet tall. The 5/8" thick shell is made by seam welding together two cylinders of stainless steel rolled plate. The base plate of the MPC is approximately 3 inches thick and the top lid is 9 inches thick. Additionally, the divider shell inside the CEC of the UMAX vault is painted with a coating developed to assist in limiting scratches to the stainless steel canister during downloading.

The canisters for the Holtec UMAX Storage System are designed and licensed to meet the stress intensity limits per ASME Section III, Subsection NB for Class 1 pressure vessels. Localized scratches are examples of local structural discontinuities per the ASME Code definition in NB-3213.3. As such, the stresses attributed to these local discontinuities are categorized as peak stresses per NB 3213.11, which are "objectionable only as a possible source of a fatigue crack or brittle facture."

Chapter 3 of the HI-STORM FSAR states that the MPC is not vulnerable to fatigue failure or brittle fracture because of the passive nature of the HI-STORM UMAX system and its highly ductile material of construction (Type 316 austenitic stainless). Namely the amplitude of cyclic stresses and pressure pulsation is limited in the pressure vessel and remains orders of magnitude below the canister's material endurance limits. Moreover, peak stresses are not subject to a prescribed stress limit as summarized in FSAR Table 2.2.10 for primary and secondary stress categories.

Therefore, FSAR Section 3.1.2.5 states failure from fatigue is not a credible concern for the HI-STORM UMAX system components. Peak stresses are specifically addressed in Table 3.1.10 of the UMAX FSAR which states: "*Increment added to primary or secondary stress by a concentration (notch), or, certain thermal stresses that may cause fatigue but not distortion. Because fatigue is not a credible source of failure in a passive system with gradual temperature changes, the cumulative damage factor from fatigue is not computed for HI-STORM UMAX components.*" The NRC inspectors concluded that

the localized scratches (peak stresses) on the canister are not a safety concern from the standpoint of ASME Section III, Subsection NB stress intensity limits.

The SONGS canisters were designed and fabricated to contain a shell thickness of 1/8" (0.125 inch) thicker than the standard canister (0.50" nominal wall thickness) associated with the Holtec UMAX Storage System. Additionally, the canisters at SONGS have been laser peened which was developed, applied, and confirmed for SONGS to add a protective layer against high tensile stress over the heat affected zones of the canister seam welds to assist in possible elimination of future stress corrosion cracking concerns. Confirmed by laboratory tests performed by the vendor and licensee, the protective layer over the welds and heat affected zones resulted in an approximately 0.080" inch (80 mil) thick layer of additional protection.

The NRC determined that scratches that occur on the surface of the MPC during insertion and withdrawal due to incidental contact with the internal features of the CEC internals are not of any safety concern from a stress limit. However, allowing the MPC to scratch, or suffer mechanical wear, presented a potential impact to the MPC design basis requirements as specified in the technical specifications. The confinement design function is required by the Holtec Certificate of Compliance 072-01040, Appendix B Technical Specifications, Section 3.3 to meet ASME Section III acceptance limits.

The ASME Section III code acceptance limits for scratches is 10 percent of the nominal wall thickness per ASME Section III, NB-3324.1 Cylindrical Shells and NB-3213.10 Local Primary Membrane Stress, which specifies a local primary membrane stress limit of 1.1Sm (or 10 percent higher than the general primary membrane stress limit). The 10 percent allowance is consistent with NUREG 2214 "Managing Aging Processes in Storage Report," Table 6-2, that states flaws must be assessed in accordance with the acceptance standards identified in ASME Section XI IWB-3514 which provides allowable flaw depths that are below 10% of nominal wall thickness.

For the 0.625-inch thick MPC shell in use at SONGS the maximum allowable scratch depth would be 0.0625 inches per ASME Section III code and required by Technical Specification 3.3, Appendix B.

The licensee performed a change under the 10 CFR 72.48 process to evaluate and accept the scratches on canisters 1 thru 29 placed in the site's UMAX ISFSI. Through the 10 CFR 72.48 process the licensee revised the FSAR Section 1.2.4 and Section 9.5 vii. design statements to allow scratches to previous and future canisters during installation and retrieval. The 10 CFR 72.48 regulation permits a licensee to make changes to the spent fuel storage cask design as described in the FSAR without obtaining prior NRC approval as long as the change does not require a change to the technical specifications or the change does not conflict with the eight criteria of 10 CFR 72.48 (c)(2).

The calculation to demonstrate the maximum depth of any possible scratch from downloading operations was documented in Holtec Dry Storage Position Paper DS-469, "Incidence and Consequence of Canister Shell Scratching from Misaligned Insertion of a Loaded MPC at SONGS," dated November 7, 2018. The DS-469 calculation was used as the basis to support a 10 CFR 72.48 evaluation performed by the licensee. Position paper DS-469 calculated the maximum force on the canister shell during downloading based on dimensional tolerances of components and the maximum angle the canister

could be misaligned. The maximum force was calculated to be approximately 2400 pound-force (lbf).

The licensee's analysis utilized Archard's wear equation to calculate the maximum depth of a possible scratch from the carbon steel shield ring to be 0.010 inches (10 mils) based on the force of 2400 lbf. The NRC inspectors reviewed the calculation and identified several inadequacies with position paper DS-469. The inadequacies included: (1) the calculation did not address contact with the harder stainless steel seismic restraints and was only based on the contact with the softer carbon steel shield ring; (2) the evaluation lacked adequate review of corrosion deposits on the stainless steel canister; and (3) the written evaluation did not address scratches and gouges in the canister's seam weld areas.

The licensee addressed the inspectors' concerns in a subsequent evaluation, HI-2188437, "Incidence and Consequence of Canister Shell Wear Scars from Misaligned Insertion of a Loaded MPC at SONGS," dated March 1, 2019. The licensee's revised 10 CFR 72.48 evaluation contained more details and analysis, which was used as a basis for concluding the change did not require prior NRC approval. The inspectors observed that evaluation HI-2188437 utilized the same methodology as the DS-469 calculation which determined the maximum depth of a possible scratch would be less than 0.0091 inches or (9.1 mils).

However, the inspectors identified additional inadequacies associated with evaluation HI-2188437 which included: (1) the licensee utilized the wrong hardness values in the calculation; (2) the hardness values did not account for the temperature of the canister; (3) the calculations utilized the wrong sling lengths for determining initial point of contact for where contact on the MPC shell could occur; and (4) the inspectors did not agree that the calculation alone could provide adequate basis without empirical evidence (i.e. testing or inspection) to support the calculation's basis.

The licensee addressed the inspectors' concerns in a revision to evaluation HI-2188437, dated March 13, 2019. Additionally, the licensee's third written evaluation included test report HI-2188450, "Simulation of High Force Contact Between MPC and UMAX CEC Storage System Components," dated March 12, 2019. In the test report, simulations were performed using representative samples for the MPC shell and UMAX CEC components most likely to damage the MPC surface. The test simulations were conducted at Holtec's Orrvilon fabrication facility. The test simulations utilized a range of test loads to demonstrate what the maximum wear on a canister would be from contact with the UMAX CEC components. Scratch depths were measured after the completion of the test runs.

The evaluation HI-2188437 calculation was revised using Archard's wear equation to contain the correct hardness values and to account for temperature of the canister. The maximum possible scratch depth utilizing the same force had decreased to 0.0024 inches (2.4 mils). However, the test data reported in test report HI-2188450 found maximum depth of scratches on the samples, using a similar test load of 2,000 lbs, to have a maximum depth of 0.007 inches (7 mils). The NRC staff concluded that the licensee test data invalidated the licensee's calculation that utilized Archard's wear equation to define the maximum possible depth of a scratch on the canister.

Subsequently, the licensee determined that the Archard's wear equation only provided an estimate of abrasive wear (removal of material from a surface by harder material) but the calculation could not account for adhesive wear (localized bonding between contacting solid surfaces leading to material transfer between two surfaces or loss from either surface). The inspectors determined that the licensee's initial written evaluations which contained numerous errors and deficiencies were inadequate and represented a violation of NRC requirements (see Section 2.2.6).

Evaluation HI-2188437 had been revised to address corrosion, pitting, and corrosion induced stress corrosion cracking (CISCC). The evaluation stated, for CISCC to occur, three conditions were necessary; a susceptible material, a strong tensile stress, and a corrosive environment. Type 316 stainless steel is a resistant austenitic material, but CISCC is possible under sufficiently severe conditions. However, for CISCC to occur, a through-wall high tensile stress is needed. The primary tensile stresses for the storage system is due to internal pressure of the helium gas which is low (approximately 45 psi). Also, the residual stresses due to rolling operations on stainless steel plates introduced a compressive stress on the outside surface of the canister shell. Seam welds of the canister were the only areas where local tensile stresses from weld shrinkage could potentially result in a through wall high tensile stress.

However, as previously explained, the canisters purchased at SONGS have been laser peened over all the seam welds and heat affected zones to provide a layer of compressive stress relief of 0.080" depth. Additionally, water is necessary for CISCC. The UMAX vault canisters are sheltered from weather intrusion. The canisters are hotter than the ambient air, so wetting from condensation is not possible during the current licensing period. Specifically, the canisters' temperature would remain above ambient temperatures well beyond the current licensing expiration date of 20 years. As such, any additional required monitoring for corrosion, pitting, and CISCC would be addressed in license renewal and through the licensee's ageing management program. The inspectors concluded that the issues related to possible corrosion, pitting, and CISCC on the canister did not pose an immediate safety concern nor immediately affect any of the system's design basis functions and could be adequately monitored and addressed as part of the licensee's ageing management program.

The licensee's subsequent written evaluation to support the site-specific 10 CFR 72.48 change to allow and bound incidental contact used in-situ visual assessment of surfaces of the canister shell and baseplate from eight loaded canisters in the UMAX ISFSI at SONGS. The sample set of eight canisters was consistent with using the guidance of ANSI ASQ Z1.4, "Sampling Procedures and Tables for Inspection by Attributes." The visual assessment was documented in "SONGS Downloading Effects on HI-STORM MPC Visual Assessment Report," dated April 15, 2019.

The eight canisters selected for inspection included: 1.) MPC serial number (SN) 067, which was involved in the August 3, 2018, misalignment incident; 2.) MPC SN 064, which was documented as having made contact with the internals of the CEC on July 22, 2018; and 3.) six additional MPCs located on different rows than the previous two MPCs. The different rows were selected to account for the drainage slope on the ISFSI pad and its potential effect on canister vertical alignment during downloading operations.

The visual assessment was performed by a robotic crawler equipped with navigational cameras and a borescope. The borescope was a flexible camera with interchangeable tips (general area tip and measurement tip). Two stages were utilized to perform the visual assessment. During the first stage, the robotic crawler and borescope with the general area tip was used to identify general locations of surface irregularities. During the section stage, the robotic crawler with the borescope using the measurement tip characterized the surface irregularities (width and depth measurements as applicable). The equipment selected by the licensee to perform the visual assessment was the General Electric borescope (VideoProbe™), along with the Robotic Technologies of Tennessee robot.

This same equipment had been used by Electric Power Research Institute for their Extended Storage Collaboration Program Non-destructive examination subcommittee, which is researching and developing technology to support inspection of dry storage canisters. This equipment had been used at multiple U.S. nuclear sites for Part 72 license renewal applications. The GE inspection Technologies' VideoProbe with Real3D™ point cloud surface scanning and analysis had been used in aviation, military, and oil & gas applications. Additionally, an NRC inspector was on-site during seven of the eight canister inspections to observe the visual assessment activities.

All surface irregularities were recorded and compared to post-fabrication photos to determine whether the surface irregularities were a result of downloading operations. All irregularities that were identified to have occurred during downloading operations were recorded and characterized. A few identified areas of interest crossed over or resided within the canisters' seam welds or weld heat affected zones. However, the protective layer of 0.080 inches provided by laser peening operations was never exceeded. The majority of wear marks identified were correlated to contact with the divider shell shield ring and had maximum wear depths of up to 0.012 inches (12 mils) deep. Additional wear marks identified were correlated to contact with seismic restraints and a maximum wear depth was 0.026 inches (26 mils) deep. Many wear marks had negligible depths.

Wear profiles for divider shell shield ring and inner seismic restraints were different. The divider shell ring wear marks were broader and shallower in comparison. The maximum depth caused by the stainless inner seismic restraint occurred over relatively short lengths in a localized narrow area and did not apply over the entire length nor width of the wear mark. In summary, the wear marks from incidental contact were not uniform, the maximum depths observed were very small in width and area and a majority of the scratch lengths contained negligible depths.

With the gathered information from the visual assessment report, the licensee performed two statistical analyses to bound the potential wear mark depths on the remaining canisters. Licensee report MPR 0299-0057-MEMO-001, "Canister Inspection Plan," dated April 15, 2019, concluded that the eight canister measurements were sufficient to support a conclusion that there is a 95 percent probability with 95 percent confidence that each of the remaining and future canisters would not have a scratch deeper than 0.035 inches (35 mils) due to downloading operations.

The second statistical analysis was documented in licensee report MPR 0299-0042-MEMO-024, "Canister Installation and Removal Effects on Wall Thickness," dated May 5, 2019. This statistical analysis determined the deepest scratch resulting from insertion and then withdrawal and assumed the two scratches occurred in

the same location. The licensee utilized the same methodology and determined that the deepest scratch at one location resulting from insertion followed by withdrawal with a 95 percent probability and 95 percent confidence to be 0.0584 inches (58 mils), which was still below the ASME code limit of 10 percent (0.0625 inches).

The NRC inspectors utilized the data obtained through the visual assessments to perform independent statistical assessments using several models that were appropriate for the sample size. The inspectors concluded, through the independent assessments, that the conclusion presented by SCE was conservative and reasonably bounded the maximum anticipated scratch or wear resulting from operational activities.

As such, the licensee's written evaluation using the visual assessments and statistical evaluations was adequate to demonstrate that the proposed change to allow the incidental contact on previous and future canisters will continue to meet the confinement design functions as specified in the FSAR and ASME Section III code tolerances and does not require a change to the storage system's technical specifications. The inspectors found that the licensee's site-specific 10 CFR 72.48 change to be acceptable and met all applicable criteria to not require NRC review and approval through a Certification of Compliance amendment.

2.2.6 Finding Related to 10 CFR 72.48 Evaluations

10 CFR 72.48(d)(1) requires, in part, that the licensee and certificate holder shall maintain records of changes in the facility or spent fuel storage cask design, of changes in procedures, and tests and experiments made pursuant to paragraph (c) of the section. These records must include a written evaluation, which provides the bases for the determination that the change does not require a Certificate of Compliance (CoC) amendment pursuant to paragraph (c)(2) of this section.

Contrary to the above, from November 7, 2018, to April 15, 2019, on two occasions the licensee did not maintain records of changes that included a written evaluation that provided the bases for the determination that the change does not require a CoC amendment pursuant to paragraph (c)(2) of 10 CFR 72.48. Specifically, the first two revisions of the 10 CFR 72.48 written evaluations to allow scratching on canisters failed to provide an adequate basis for determination that the change did not require a CoC amendment. As noted in Section 2.2.5.b of this report, the inspectors identified numerous technical errors with the calculations used as the bases for the 10 CFR 72.48 written evaluations. In addition, the first two revisions of the licensee's written evaluation did not demonstrate that the maximum possible scratch depth would not exceed ASME Section III code limits, a technical specification requirement.

The inspectors determined that the finding was of low safety significance because the inspectors assessed that the in-situ visual assessment and statistical analysis provided an adequate basis for the determination that the canister will continue to meet structural and confinement design functions as specified in the FSAR and continue to meet ASME Section III code tolerances.

The inspectors determined that the violation was similar to the violation examples in Section 2.1.3.D.5 of the NRC Enforcement Manual, which states that violations of 10 CFR 50.59 will be considered more than minor and categorized at Severity Level IV if

the licensee failed to perform an adequate 10 CFR 72.48 evaluation, similar to a 10 CFR 50.59 evaluation, that resulted in a condition having low safety significance.

Consistent with the guidance in Section 1.2.6.D of the NRC Enforcement Manual, if a violation does not fit an example in the enforcement policy violation examples, it should be assigned a severity level: (1) commensurate with its safety significance, and (2) informed by similar violations addressed in the violation examples. The violation was evaluated to be similar to Enforcement Policy Section 6.1.d.2.

The licensee entered the finding into the CAP as AR 1218-11302 and AR 0219-96601. The licensee restored compliance by revising the written evaluation to provide an adequate basis to conclude the change did not require NRC approval. Specifically, the revised written evaluation provided a basis that incidental contact of the canister with the internal components of the CEC during insertion and withdrawal operations would not remove greater than 10% nominal wall thickness of the canister in accordance with ASME Section III which was required by Appendix B Technical Specification 3.3 requirements. Because the licensee entered the issue into the CAP, the safety significance of the issue was low, and the issue was not repetitive or willful, this Severity Level IV violation was treated as a NCV, consistent with Section 2.3.2.a of the NRC Enforcement Policy (NCV 07200044/2018-002-04, Failure to provide adequate written basis for 72.48 change (10 CFR 72.48)).

2.2.7 <u>(Closed) Unresolved Item 07200041/2017-001-02, 10 CFR 72.48 Methodology</u>

NRC Inspection Report 05000206/2017-003, 05000361/2017-003, 05000362/2017-003, and 07200041/2017-001 dated, August 24, 2018 (ADAMS Accession ML18200A400), documented an Unresolved Item (URI) 07200041/2017-001-02, "10 CFR 72.48 Methodology." The issue related to a 10 CFR 72.48 evaluation for the scenario of a hypothetical accident of the loaded HI-TRAC VW transfer cask contacting the sides and bottom of the spent fuel pool, which was analyzed in report HI-2177713 "HI-TRAC VW Drop in Cask Storage Pool at SONGS," Revision 1.

For a short period of time, the HI-TRAC VW and loaded MPC was in an unconstrained condition on an intermediate shelf in the spent fuel pool. If a DBE seismic event was to occur during that time frame, the HI-TRAC VW with a loaded MPC could hypothetically fall to the lower level of the spent fuel pool and experience a higher lateral force than previously analyzed by the HI-STORM FW and UMAX FSARs. In report HI-2177713, the licensee demonstrated acceptability of the peak impact deceleration for the HI-TRAC VW scenario at SONGS by comparing those lateral forces to the peak impact deceleration values used to support the 10 CFR Part 71 HI-STAR 190 transport package safety analyses which utilized the same canister.

The licensee's evaluation concluded that the maximum peak lateral deceleration value of the HI-TRAC VW in the pool at SONGS to be 74g's, which was below the HI-STAR 190 side drop evaluation of 85.9g's. Additionally, the MPC and fuel basket evaluated stresses were identified by the licensee to be less than the design basis criteria described in the limiting values from HI-STORM FW FSAR, Section 2.2.8. The licensee stated that the same computer software (LS-DYNA) was utilized in all three evaluations (SONGS site-specific drop evaluation, HI-STORM FW/UMAX FSAR non-mechanistic tip-over evaluation, and HI-STAR FSAR transportation cask drop evaluation).

At the time of the initial inspection, the NRC needed more information to determine if the utilization of evaluations conducted for the 10 CFR Part 71 HI-STAR 190 transportation license to bound conditions for storage operations under 10 CFR Part 72 UMAX license through SONGS's 10 CFR 72.48 process was appropriate and in compliance with NRC regulations. The NRC subsequently determined that licensee's change was in violation of 10 CFR 72.48 requirements.

The UMAX FSAR references the FW FSAR for the use of the HI-TRAC VW, also both FSARs discuss various tip-over/drop events or requirements that must be followed such that a tip-over/drop event is not credible.

The FW FSAR, Table 1.2.10, "Criteria for Site-Specific Safety Qualification of HI-TRAC VW," item #10 states, in part, *the transfer cask's kinematic stability is established under all loading evolutions where the cask is freestanding to ensure kinematic compliance (no tip-over or collision with a proximate structure).*

Additionally, a tip-over/drop event as well as kinematic stability of a canister in a HI-TRAC VW was described as either a non-credible accident or must be demonstrated per analysis to have kinematic stability for tornado missiles (FW Section 2.2.3 e.), cask handling (FW Section 2.2.3 f.), and transportation operations (UMAX Appendix B, Technical Specification 3.4.15).

Nuclear Energy Institute Guidance Document 96-07, Appendix B, "Guidelines for 10 CFR 72.48 Implementation," Section 4.3.5, states that, "a change or activity, which increases the frequency of an accident previously thought to be incredible to the point where it becomes as likely as the accidents in the FSAR, could create the possibility of an accident of a different type."

10 CFR 72.48 (c)(1)(ii)(C) states in part, a licensee may make a change in the facility or spent fuel storage cask design as described in the FSAR without obtaining a CoC amendment if the change does not meet any of the criteria in paragraph (c)(2).

10 CFR 72.48 (c)(2)(v) states in part, a general licensee shall request that the certificate holder obtain a CoC amendment pursuant to 10 CFR 72.244, prior to implementing a proposed change if the change would: Create a possibility for an accident of a different type than any previously evaluated in the FSAR.

Contrary to the above, from January 30, 2018, to August 3, 2018, the licensee made a change in the spent fuel storage cask design as described in the FSAR and failed to request the certificate holder to obtain a CoC amendment prior to implementing the proposed change which created a possibility of an accident of a different type than any previously evaluated in the FSAR. Specifically, the licensee created the possibility of a new accident not previously analyzed in the FSAR through a 10 CFR 72.48 change (10 CFR 72.48 Assignment 0718-10512-3) to allow placement of a loaded HI-TRAC VW cask on an intermediate shelf in the spent fuel pool which was evaluated, by the licensee, to not be kinematically stable and had the potential to collide with proximate structures during a seismic event.

This violation was dispositioned per the traditional enforcement process using Section 2.3 of the NRC's Enforcement Policy. The inspectors determined that the finding was of low safety significance since the accident condition of a spent fuel cask

drop (due to a seismic event) from the intermediate shelf in the cask pool to the lower portion of the cask pool was an accident condition that had been analyzed and NRC approved in NUREG-0712, "Safety Evaluation Report related to the operation of SONGS Units 2 and 3, dated February 1981," and described in the SONGS Decommissioning Safety Analysis Report Section 15.1.1.5. Additionally, the licensee's calculations demonstrated that maximum lateral deflection in the fuel basket's active fuel region would not have exceeded requirements in the Holtec FW FSAR.

The inspectors determined that the violation was similar to the violation examples in Section 2.1.3.D.5 of the NRC Enforcement Manual, which states that violations of 10 CFR 50.59 will be considered more than minor and categorized at Severity Level IV if the licensee failed to request a license amendment, the NRC would likely approve the amendment, and the change resulted in a condition having low safety significance.

Consistent with the guidance in Section 1.2.6.D of the NRC Enforcement Manual, if a violation does not fit an example in the Enforcement Policy Violation Examples, it should be assigned a severity level: (1) commensurate with its safety significance; and (2) informed by similar violations addressed in the Violation Examples. The violation was evaluated to be similar to Enforcement Policy Section 6.1.d.2

The licensee entered the issue into the CAP as AR 0718-10512 and AR 0617-86918. The licensee restored compliance by revising the loading procedures to no longer utilize the intermediate shelf in the pool. The revised procedures required the transfer cask to be moved, after spent fuel assembly loading, from the bottom of the spent fuel pool directly to the cask wash-down pit for further processing (see Section 2.2.8). Because the licensee entered the issue into the CAP, the safety significance of the issue was low, and the issue was not repetitive or willful, this Severity Level IV violation was treated as a NCV, consistent with Section 2.3.2.a of the Enforcement Policy (NCV 07200044/2018-002-05, Failure to request the certificate holder to obtain a CoC amendment (10 CFR 72.48)).

No additional deficiencies were identified during the review of the Unresolved Item. This Unresolved Item 07200041/2017-001-02, "10 CFR 72.48 Methodology," is closed.

2.2.8 Dry Runs (Transportation, Downloading, Uploading)

Week of January 28, 2019

During the week of January 28, 2019, inspectors observed SCE perform demonstrations of sections of revised procedures HPP-2464-400, "MPC Transfer at SONGS," Revision 19 and HPP-2464-500, "MPC Unloading at SONGS," Revision 6. The demonstrations for this week of NRC on-site inspection activity involved movement of the HI-TRAC VW transfer cask with a canister simulator from the Unit 2 fuel building along the haul path to the ISFSI pad and included downloading operations.

During the first day of field demonstrations, SCE demonstrated spent fuel travel along a revised travel path for the low-profile transporter while carrying the canister simulator and HI-TRAC VW transfer cask from the Unit 2 fuel building. The haul path was revised based on seismic analyses and the revisions were intended to keep the low-profile transporter and transfer cask the required height and distance from structures along the path that could possibly be impacted if a seismic event were to occur during travel. The

revised path included white and yellow painted lines on the pavement to serve as guides for the operator to travel within. There were also restricted zone markings on the haul path near adjacent structures that were required to be avoided. The transfer cask was transported by the operator from the fuel building to the outside of the plant protected area, and into the SONGS ISFSI protected area, where it met up with the VCT. The VCT continued the movement of the canister simulator onto the ISFSI pad and into stack-up configuration for downloading.

The transfer cask was transported by use of the VCT until it was secured to the UMAX ISFSI mating device. A nighttime downloading demonstration of the canister simulator was performed after the ISFSI haul path travel demonstration. No adverse conditions were identified during the downloading demonstration operations. The new load monitoring equipment, cameras, and personnel present on the ISFSI pad ensured that loss-of-load indications was promptly responded to during downloading operations. The new equipment worked as intended and provided a positive load indication for the canister simulator. The cask loading crew used procedure adherence and the equipment enhances at their disposal to successfully perform the nighttime downloading demonstration.

The following day, the cask loading crew used the most recent revision of procedure HPP-2464-500 to demonstrate removal of the simulator from the UMAX ISFSI vault. Uploading operations proceeded without any issues. In the same manner as the previous evening, the cask loading crew used procedure adherence and the equipment enhancements at their disposal to successfully retrieve the canister simulator from the ISFSI vault.

Finally, a daytime downloading operation was demonstrated in accordance with procedure HPP-2464-400. The daytime downloading proceeded with the same requirements as the nighttime demonstration. The inspectors observed rigorous procedure adherence and oversight supervision during the cask loading operations.

Week of February 11, 2019

During the week of February 11, 2019, NRC observed SCE perform demonstrations of sections of its revised procedures HPP-2464-400, "MPC Transfer at SONGS," Revision 19, and HPP-2464-500, "MPC Unloading at SONGS," Revision 6, inside the fuel building. The second-week demonstrations were performed to support procedure revisions that removed usage of the spent fuel pool intermediate shelf location during fuel loading operations. To remove usage of the intermediate shelf required that the crane hook be fully immersed into the pool when placing the transfer cask and empty canister into the cask loading pit. The previous procedure revision avoided immersing the crane hook, block, and wire rope into the potentially contaminated spent fuel pool water.

To facilitate the procedure revisions, SCE performed modifications to the Unit 2 cask handling crane hook that would allow it to be immersed into the spent fuel pool water. At the time of the inspection, the Unit 3 cask handling crane hook had not yet been modified. However, the inspectors noted that the work orders were in place for the modification.

The inspectors observed SCE successfully demonstrate placement of an empty transfer cask and canister into the spent fuel cask loading pit. Next, the licensee successfully demonstrated placement of the MPC lid and drain tube into the transfer cask while at the bottom of the cask loading pit and removal of the transfer cask from the cask loading pit to the cask washdown area. The inspectors observed rigorous procedure adherence and oversight supervision during the fuel loading operations.

2.2.9 (Closed) Notice of Violation SLII 072-00041/2018-001-02, "Failure to ensure redundant drop protection features were available" (10 CFR 72.212), EA-18-155

As a result of the NRC Special Inspection a violation was identified for the licensee's failure to provide redundant drop protection features during downloading operations.

The licensee submitted its response to the NRC letter within the required 30-day time frame, on April 23, 2019 (ADAMS Accession ML19116A056), which contained the corrective steps taken to ensure full compliance was achieved.

During supplemental inspection activities conducted from November 2018 to May 2019, the NRC inspectors concluded that SCE's proposed and completed corrective actions, as described in this report, restored compliance, addressed extent of condition, and were adequate to prevent recurrence. No additional deficiencies were identified during NRC's review of this violation.

This closes VIO 072-00041/2018-001-02, "Failure to ensure redundant drop protection features are available," (10 CFR 72.212), EA-18-155.

2.2.10 ISFSI Pad Surveys

On October 22, 2018, during a routine decommissioning inspection (ADAMS Accession ML18323A024) the NRC inspectors performed independent measurements and verifications of the radiological conditions at the SONGS ISFSI. The inspectors measured various locations including the background areas, public access areas, owner-controlled areas, protected areas, and representative locations on both generally licensed ISFSI Pads: Transnuclear, (TN) Inc. Nuclear Horizontal Modular Storage (NUHOMS) and Holtec HI-STORM UMAX dry fuel storage systems.

The inspectors used a Ludlum Model 19, NRC Tag Number 033906, serial number 84259 with a calibration due date of July 23, 2019, to perform the survey measurements. The data in Attachment 2 shows the ranges of the measurements of each UMAX location by the VVM number at the inlet air vents, closure lid, and outlet air vent. Attachment 2, also shows the measurements taken on the NUHOMS locations, on contact with the inlet vent and 1 foot away from the inlet vent.

The VVM with the highest gamma measurement was VVM 33 with the inlet air vents ranging from 310-330 μR/hr. The NUHOMS location with the highest gamma measurement was TN 21, on contact with the inlet vent was 1,600 μR/hr. Background measurements from around the site ranged from 3-10 μR/hr. The NRC inspectors did not identify any measurements at the owner-controlled area boundary or in the public access areas to be above normal background measurements. A more detailed discussion of the surveys taken can be found at "NRC Surveys of SONGS ISFSI Pad,"

dated October 22, 2018 (ADAMS Accession ML19011A457) and on the provided table in Attachment 2 of this report.

2.3 Conclusions

The inspectors reviewed two LERs and one licensee event notification which had been reported to the NRC since the last inspection. The review of the event notification resulted in one Severity Level IV violation of NRC requirements that was treated as a NCV. The inspectors reviewed inspection follow-up items from the NRC Special Inspection Report which included the NRC's evaluation of the licensee's drop analysis, scratch analysis, and observations of dry run demonstrations. The review of the scratch analysis resulted in one Severity Level IV violation of NRC requirements that was treated as a NCV. The inspectors closed one violation which resulted from the NRC Special Inspection for the licensee's failure to ensure redundant drop protection features during downloading operations on August 3, 2018. The inspectors documented the results of the independent measurements and verifications of the radiological conditions at the SONGS ISFSI.

3 Exit Meeting Summary

On February 15, 2019, following an onsite portion of the inspection, the inspectors provided a debrief of the preliminary results to Mr. Doug Bauder, Vice President and Chief Nuclear Officer, and other members of the licensee staff. The licensee acknowledged the issues presented by the NRC inspection team.

On March 25, 2019, the NRC performed a public webinar meeting to discuss the inspection team's preliminary results. On March 28, 2019, the NRC participated in a San Onofre Community Engagement Panel Meeting to discuss the inspection team's preliminary results. On June 3, 2019, the NRC performed a public webinar meeting to discuss the NRC's decision on resumption of fuel loading activities at SONGS. On June 5, 2019, the NRC participated in a San Onofre Community Engagement Panel Meeting and discussed the NRC's decision on resumption of fuel loading activities at SONGS.

On June 13, 2019, the inspectors presented the final inspection results to Mr. Al Bates, Regulatory and Oversight Manager and other members of the licensee staff. The licensee acknowledged the issues presented.

SUPPLEMENTAL INSPECTION INFORMATION

PARTIAL LIST OF PERSONS CONTACTED

Licensee Personnel

A. Bates, Regulatory and Oversight Manager
M. Morgan, Regulatory and Oversight
L. Bosch, Plant Manager
T. Palmisano, former Vice President Decommissioning and Chief Nuclear Officer
J. Pugh, Project Engineer
K. Rod, General Manager Decommissioning Oversight
J. Smith, Project Manager, Holtec
M. Soler, Vice President Quality, Holtec

INSPECTION PROCEDURES USED

IP 92702	Follow-up on Traditional Enforcement Actions
IP 71153	Follow-up of Events and Notices of Enforcement

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened and Closed

07200044/2018-002-01	NCV	Failure to ensure appropriate quality standards (10 CFR 72.146)
07200044/2018-002-02	NCV	Failure to ensure purchased material conformed to the procurement documents (10 CFR 72.154)
07200044/2018-002-03	NCV	Failure to ensure the loaded transfer cask and its conveyance was evaluated under the site-specific DBE (10 CFR 212)
07200044/2018-002-04	NCV	Failure to provide adequate written basis for 72.48 change (10 CFR 72.48)
07200044/2018-002-05	NCV	Failure to request the certificate holder to obtain a CoC amendment (10 CFR 72.48)

Closed

072-00041/2018-001-01	VIO	Failure to identify and correct conditions adverse to quality (10 CFR 72.172) EA-18-155
072-00041/2018-001-02	VIO	Failure to ensure redundant drop protection features were available (10 CFR 72.212) EA-18-155

072-00041/2018-001-03	VIO	Failure to assure that operations of important to safety equipment were limited to trained and certified personnel (10 CFR 72.190) EA-18-155
072-00041/2018-001-04	VIO	Failure to provide adequate instructions or procedures (10 CFR 72.150) EA-18-155
072-00041/2018-001-05	VIO	Failure to make 24-hour notification (10 CFR 72.75) EA-18-155
2018-001-0	LER	Spent Nuclear Fuel Canister Temporarily Wedged in Dry Cask Storage Container
53858	EN	Inadequate Analysis for VCT Operations
07200041/2017-001-02	URI	10 CFR 72.48 Methodology
Discussed		
2018-002-0	LER	Spent Nuclear Fuel Transport Conveyance Vehicle Operated Outside Obstacle Clearance Limit

LIST OF ACRONYMS USED

ACE	Apparent Cause Evaluation
ADAMS	Agencywide Documents Access and Management System
AHSM	Advanced Horizontal Storage Module
ASME	American Society of Mechanical Engineers
AR	Action Request
ASME	American Society of Mechanical Engineers
AV	Apparent Violation
CA	Corrective Action
CAP	Corrective Action Program
CAPR	Corrective Action to Prevent Recurrence
CCE	Common Cause Evaluation
CEC	Cavity Enclosure Container
CFR	*Code of Federal Regulations*
CISSC	corrosion induced stress corrosion cracking
CoC	Certificate of Compliance
DBE	Design Basis Earthquake
EN	Event Notification
FCR	Field Condition Report
FSAR	Final Safety Analysis Report
GTCC	Greater than Class C
HI-STORM FW	Holtec International Storage Module Underground Flood and Wind
HI-STORM UMAX	Holtec International Storage Module Underground Maximum Capacity
IP	Inspection Procedure
ISFSI	Independent Spent Fuel Storage Installation
ITS	Important-to-Safety
LER	Licensee Event Report
NECP	Nuclear Engineering Change Package
NCV	Non-Cited Violation
NITS	Not-Important-to-Safety
NRC	U.S. Nuclear Regulatory Commission
NUHOMS	Nuclear Horizontal Modular Storage
MPC	multipurpose canister
QI	Quality Investigation
RCE	Root Cause Evaluation
RRCE	Reportability Root Cause Evaluation
SAT	Systematic Approach to Training
SCE	Southern California Edison
SL	Severity Level
SONGS	San Onofre Nuclear Generating Station
TN	Transnuclear
VCT	Vertical Cask Transporter
VIO	Violation
VVM	Vertical Ventilated Module or vault

Radiological Surveys of ISFSI Pads

Table 1, Holtec HI-STORM UMAX ISFSI Pad Survey Results

Vertical Ventilated Module	Inlet Air Vent Range (µR/hr)	Closure Lid Range (µR/hr)	Outlet Air Vent Range (µR/hr)
22	130-160	9-15	110-120
23	170-230	12-17	150-160
24	180-240	11-14	150-170
25	210-240	11-17	170-190
26	180-230	11-16	130-140
27	160-220	9-17	140-160
28	230-300	14-19	210-220
29	200-320	13-18	190-210
30	190-280	12-19	180-190
31	190-220	13-19	170-180
32	200-260	13-18	170-190
33	310-330	13-18	230-240
44	220-260	14-21	180-200
45	180-250	14-20	190-210
46	270-320	15-22	220-240
47	180-250	11-20	170-180
58	130-180	11-17	120-160
59	150-200	14-20	130-150
60	170-200	15-19	140-160
61	160-200	11-18	140-150
67	140-210	11-17	140-150
68	120-160	11-16	130-140
69	160-210	11-16	140-160
70	180-210	13-18	140-150
71	190-220	11-17	140-160
72	120-190	11-15	140-160
73	180-220	11-17	150-170
74	160-180	11-16	130-160
75	100-260	11-16	180-210

Table 2, TN, Inc. NUHOMS ISFSI Pad Survey Results

AHSM	Inlet Vent Contact (µR/hr)	Inlet Vent 1 Foot Away (µR/hr)
1	800	500
2	700	500
3	800	500
4	800	500
5	700	500
6	700	500
7	600	400
8	700	500

Attachment 2

AHSM	Inlet Vent Contact (µR/hr)	Inlet Vent 1 Foot Away (µR/hr)
9	700	500
10	600	400
11	800	500
12	700	500
13	600	400
14	500	300
15	100	70
16	420	260
17	440	240
18	440	270
19	1400	900
20	1300	1000
21	1600	1100
22	1000	700
23	1000	700
24	900	600
25	600	400
26	380	220
27	1000	600
28	800	600
29	1000	700
30	1200	800
31	800	500
32	1200	700
33	900	500
34	1100	800
35	900	500
36	1100	700
37	1000	600
38	1200	800
39	1000	600
40	1100	700
41	1100	700
42	1100	700
43	320	180
44	320	180
45	310	170
46	310	210
47	310	180
48	900	600
49	700	500
50	360	210
51	360	220

UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION IV
1600 E. LAMAR BLVD
ARLINGTON TX 76011-4511

August 24, 2018

Mr. Thomas J. Palmisano
Vice President and Chief Nuclear Officer
Southern California Edison Company
San Onofre Nuclear Generating Station (SONGS)
P.O. Box 128
San Clemente, CA 92674-012

SUBJECT: SAN ONOFRE NUCLEAR GENERATING STATION – NRC INSPECTION REPORT 05000206/2017-003, 05000361/2017-003, 05000362/2017-003, AND 07200041/2017-001

Dear Mr. Palmisano:

This letter refers to routine U.S. Nuclear Regulatory Commission (NRC) team inspections conducted from June 2017 through June 2018. The purpose of the inspection was to observe your dry fuel storage preoperational testing activities, to independently assess your readiness to load spent fuel into the newly constructed UMAX Independent Spent Fuel Storage Installation (ISFSI), and to inspect initial fuel loading operations. The initial loading of the spent fuel into the first dry fuel storage cask of your UMAX ISFSI occurred between January 22-31, 2018. After continued in-office review of information following the loading of the first canister into the UMAX ISFSI, a final telephonic exit meeting was conducted on August 8, 2018, with Mr. Lou Bosch, Plant Manager, and other members of your staff.

The NRC inspection team examined activities conducted under your license as they relate to public health and safety, and to confirm compliance with the Commission's rules and regulations, and with the conditions of your license. The inspection reviewed compliance with the requirements specified in the Holtec HI-STORM UMAX storage system's Certificate of Compliance 72-1040, the associated Technical Specifications, the FW and UMAX Final Safety Analysis Reports, and the regulations in Title 10 of the *Code of Federal Regulations* (CFR) Parts 20, 50, and 72. Within these areas, the inspection consisted of selected examination of procedures and representative records, observations of activities, and interviews with personnel. The inspection determined that you had completed all required activities identified in the Holtec Certificate of Compliance 72-1040 for use of the Holtec HI-STORM UMAX storage system at your site.

Based on the results of these inspections, the NRC has determined that one Severity Level IV violation of NRC requirements occurred. The violation was related to the design control of field changes made to important to safety equipment associated with your loading activities. Because the violation was of low safety significance and the licensee initiated a condition report with appropriate resolutions to address and correct the issue, this violation is being treated as a Noncited Violation (NCV), consistent with Section 2.3.2 of the NRC Enforcement Policy. The NCV is described in the subject inspection report.

Additionally, the NRC opened an Unresolved Item (URI) related to the methodology utilized in the licensee's 10 CFR 72.48 evaluation regarding a hypothetical transfer cask drop within the spent fuel pool during a seismic event. Additional information is needed to determine if the change could be performed through the 10 CFR 72.48 process. The URI is described in the subject inspection report.

If you contest the violation or significance of the NCV, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington DC 20555-0001, with copies to: (1) the Regional Administrator, Region IV and (2) the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, DC 20555-0001.

In accordance with 10 CFR 2.390 of the NRC's "Agency Rules of Practice and Procedure," a copy of this letter, its enclosure, and your response, if you choose to provide one, will be made available electronically for public inspection in the NRC Public Document Room or from the NRC's Agencywide Documents Access and Management System, accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html. To the extent possible, your response should not include any personal, privacy, or proprietary information so that it can be made available to the public without redaction.

Should you have any questions concerning this inspection, please contact the undersigned at (817) 200-1151 or Mr. Lee Brookhart at (817) 200-1549.

Sincerely,

/RA/

Janine F. Katanic, PhD, CHP, Chief
Fuel Cycle and Decommissioning Branch
Division of Nuclear Materials Safety

Dockets: 50-206; 50-361; 50-362; 72-041
Licenses: DPR-12; NPF-10; NPF-15

Enclosure:
Inspection Report 05000206/2017003, 05000361/2017003, 05000362/2017003, and 07200041/2017001

w/attachments:
Supplemental Information

U.S. NUCLEAR REGULATORY COMMISSION
REGION IV

Dockets:	05000206; 05000361; 05000362; 07200041
Licenses:	DPR-13; NPF-10; NPF-15
Report Nos.:	05000206/2017-003; 05000361/2017-003; 05000362/2017-003; 07200041/2017-001
Licensee:	Southern California Edison Company (SCE)
Facility:	San Onofre Nuclear Generating Station, Units 1, 2, 3 and Independent Spent Fuel Storage Installation
Location:	5000 South Pacific Coast Highway, San Clemente, California
Inspection Dates:	June 26-30, 2017, Welding Dry Run Demonstration August 1-3, 2017, Fluid Operations Dry Run Demonstration September 25-28, 2017, Transporter Heavy Loads Demonstration October 9-13, 2017, Programs Review December 4-7, 2017, Fuel Building Heavy Loads Demonstration January 22-31, 2018, First Canister Loading Operation
Inspectors:	Lee Brookhart, Senior Inspector Fuel Cycle and Decommissioning Branch Eric Simpson, Inspector Fuel Cycle and Decommissioning Branch Marlone Davis, Senior Transportation and Safety Inspector Inspections and Operations Branch NMSS, Division of Spent Fuel Management Earl Love, Senior Transportation and Safety Inspector Inspections and Operations Branch NMSS, Division of Spent Fuel Management
Approved By:	Janine F. Katanic, PhD, CHP, Chief Fuel Cycle and Decommissioning Branch Division of Nuclear Materials Safety

Enclosure

EXECUTIVE SUMMARY

San Onofre Nuclear Generating Station, Units 1, 2, 3, and ISFSI
NRC Inspection Report 05000206/2017003; 05000361/2017003; 05000362/2017003; 07200041/2017001

Between June 2017 and January 2018, the NRC conducted six separate on-site inspections related to the San Onofre Nuclear Generating Station's (SONGS) program for the safe handling and storage of spent fuel at their UMAX Independent Spent Fuel Storage Installation (ISFSI). The inspection teams observed five dry run pre-operational training demonstrations and the loading of the first spent fuel canister for the Holtec UMAX cask system. The licensee selected the Holtec Certificate of Compliance No. 72-1040, HI-STORM UMAX cask storage system to house the remaining fuel from Units 2 and 3 after the decision was made to cease power operations. The ISFSI was licensed by the NRC under the general license provisions of Title 10 *Code of Federal Regulations* (CFR) Part 72, Subpart K.

Topical areas reviewed during the inspections included overhead crane requirements, loading operations, fuel verification, radiation protection, quality assurance, nondestructive testing, training, welding, and fire protection. Between the site dry run inspections and continuing after the first loading inspection, an in-office review was performed by the NRC inspectors relating to additional documentation provided by the SONGS staff. This effort involved the review of licensee reports, procedures, calculations, training documentation, test results, personnel qualification records, safety evaluations, and condition reports. During the dry run inspections, the licensee completed the pre-operational demonstrations of equipment and the implementation of the procedures to verify all operations required by the conditions of the license and the technical specifications could be performed safely. The first cask was placed within the SONGS UMAX ISFSI on January 31, 2018.

<u>Preoperational Testing of an ISFSI (60854)</u>

- Forced helium dehydration dryness limits, helium purity, and helium backfill requirements had been incorporated into the licensee's procedures. Operation of the forced helium dehydration system and backfill to the required dryness limits was demonstrated during the pre-operational dry run exercises and first loading activities. (Section 1.2.a)

- The cask loading cranes used in the spent fuel handling buildings to lift the spent fuel canisters had been previously accepted by the NRC as single failure proof cranes. The cranes were designed to retain control of and hold loads during design basis seismic events at the SONGS site. Calculations were reviewed by NRC's Division of Spent Fuel Management that demonstrated that the forces from a seismic event in the upward and horizontal directions would not exceed the strength of the crane's seismic restraints. Additional seismic evaluations were reviewed to ensure seismic stability during transfer operations. This review included the transfer cask (loaded with a canister) in the spent fuel building during decontamination and closure operations, on the low profile transporter, on the vertical cask transporter, and during transfer of the canister into the UMAX ISFSI. Based on the review of the design documents and calculations, the Division of Spent Fuel Management's staff concluded that there was reasonable

assurance that the cranes and other handling/restraining equipment were structurally adequate to withstand design basis earthquake loads during fuel loading operations. (Section 1.2.b)

- The 125-ton spent fuel building cranes were subjected to daily prior-to-use inspections that satisfied the requirements of American Society of Mechanical Engineers (ASME) B30.2, "Overhead and Gantry Cranes". On an annual basis the cranes were subjected to a more rigorous inspection that met the requirements of ASME B30.2 and the Ederer Generic Licensing Topical Report EDR-I(P) "Ederer's Nuclear Safety Related Extra Safety and Monitoring Cranes," Revision 3. (Section 1.2.c)

- The 125-ton spent fuel building cranes were properly load tested, as required by ASME B30.2, in the fall of 2017. The tests included a full performance test with 100 percent of the maximum critical load and a 125 percent static load test. The cranes' hooks were subjected to a 200 percent hook load test in 2003 by Ederer Inc. (Section 1.2.d)

- The NRC inspectors observed the licensee successfully complete all the required pre-operational tests specified in the Certificate of Compliance. This included fuel assembly selection, welding, nondestructive testing, drying, helium backfilling, and the unloading of a sealed canister. A weighted canister was used to demonstrate heavy load activities inside the fuel handling building, transport between the fuel handling building and the ISFSI, and movement back into the fuel handling building for unloading purposes. (Section 1.2.e)

- The licensee's fuel loading characterization plan met the Certificate of Compliance limits for length, width, weight, irradiation cooling time, average burn-up, cladding, decay heat, and fuel enrichment. The licensee had established provisions for independent verification of the correct loading of spent fuel assemblies into the canister. (Section 1.2.f)

- The licensee had incorporated the requirements related to heavy loads for lift height limits, travel paths, and temperature restrictions during movement of the transfer cask into its procedures. The site's vertical cask transporters were load tested and maintained in accordance with NUREG-0612 criteria. (Section 1.2.g)

- The requirements for nondestructive testing of a spent fuel canister were incorporated into the licensee's procedures. The helium leak testing equipment used during the dry run demonstration and first loading was verified to meet the requirements listed in the technical specifications. The visual and liquid dye penetrant examination procedures implemented all the applicable requirements from ASME Boiler and Pressure Vessel Code Section III, Section IV, and the Final Safety Analysis Report regarding nondestructive examination of welds. (Section 1.2.h)

- The requirements for canister hydrostatic testing had been incorporated into the licensee's procedures and were consistent with the requirements of ASME Boiler and Pressure Vessel Code Section III Subsection NB, Article NB-6000. The hydrostatic testing sequence and criteria described in the Final Safety Analysis Report had been incorporated into the licensee's procedures. (Section 1.2.i)

- The licensee's special lifting device program complied with American National Standard Institute (ANSI) N14.6, "Special Lifting Devices for Shipping Containers Weighing 10,000 Pounds or More," (1993) criteria for stress design, annual inspections, and 300 percent proof loadings for the MPC lift cleats, HI-TRAC lift lugs, HI-TRAC lift links, lift yokes, and the lift yoke extensions. (Section 1.2.j)

- The licensee had established procedures and work orders to perform the required daily monitoring surveillances required by the technical specifications, monthly vent inspections for damage, and monthly/annual/five year inspections of the ISFSI and Vertical Ventilated Module per Final Safety Analysis Report requirements. (Section 1.2.k)

- All welding procedures contained the required variables specified in ASME Boiler and Pressure Vessel Code Section IX for gas tungsten arc welding. Requirements for hydrogen monitoring during welding of the inner cask lid had been incorporated into the procedures. The welders had met the qualification testing requirements for manual and machine welding of the canister lid. (Section 1.2.l)

Operations of an ISFSI (60855)

- The first loading inspection conducted in January 2018 included 24-hour observation of loading operations for the critical tasks associated with the licensee's first UMAX loading. Inspectors observed operations which included fuel loading, heavy lifts associated with the fuel building crane, welding and nondestructive testing of the canister lid-to-shell weld, hydrostatic pressure testing, forced helium dehydration, helium backfill, vent/drain port cover welding and nondestructive testing, helium leak testing, radiological surveying, and transport of the loaded transfer cask to the UMAX ISFSI pad. (Section 2.2.a)

- During the first loading operations, the NRC inspectors identified one violation of 10 CFR 72.146 (c), "Design Control," requirements. The licensee had made modifications to Important to Safety components associated with the transfer cask seismic restraint system through the vendor's (Holtec) corrective action program and did not follow the SONGS Engineering Design Change Process. The licensee failed to ensure that design changes or field changes to Important to Safety components were subjected to design control measures commensurate with those applied to the original design. The original documentation for the changes did not contain a rigorous engineering analysis that demonstrated the changes were acceptable and those changes were not properly accepted for implementation through the Licensee's 10 CFR 50.59/72.48 program. This violation was determined to have a low safety significance since all the deviations or modifications from the original design were subsequently found to be acceptable and the changes did not affect the specific components' safety design function or bases. Because the licensee entered the issue into their corrective action program, the safety significance of the issue was low, the licensee restored compliance, and the issue was not found to be repetitive or willful, this Severity Level IV violation was treated as a Noncited Violation, consistent with the NRC Enforcement Policy. (Section 2.2.b)

Review of 10 CFR 72.212(b) Evaluations (60856)

- Emergency planning provisions for the UMAX ISFSI had been incorporated into the site's emergency plan. This included adding a specific emergency action level for an event involving damage to a loaded UMAX casks. (Section 3.2.a)

- A fire and explosion hazards analysis had been performed specific to the SONGS UMAX ISFSI. Administrative controls were established to limit the quantity of combustible and flammable liquids around the ISFSI and near the transport path during movement of the canister. The licensee provided calculations demonstrating that the worst case postulated fire event during transportation would not result in a significant increase in the temperature of the spent fuel inside a loaded canister. (Section 3.2.b)

- The licensee evaluated the bounding environmental conditions specified in the Holtec Final Safety Analysis Report and Certificate of Compliance 72-1040 Technical Specifications against actual site conditions. These included: tornados/high winds, flood, seismic events, tsunamis, hurricanes, lightning, burial of the ISFSI under debris, normal and abnormal temperatures, collapse of nearby facilities, and fires/explosions. The site environmental conditions at SONGS were bounded by the Holtec storage system's design parameters. (Section 3.2.c)

- The licensee had implemented its approved reactor facility 10 CFR Part 50 quality assurance program and corrective action program for the activities associated with the UMAX ISFSI. Selected quality assurance activities were reviewed related to calibrations, audits, surveillances, and receipt inspections. (Section 3.2.d)

- The licensee had incorporated keeping radiation exposures As Low as Reasonably Achievable into planning for the cask loading program. Requirements for radiation surveys described in the Final Safety Analysis Report and technical specifications had been incorporated into the licensee's procedures for cask loading operations. Projected radiation levels at the ISFSI were calculated for an assumed individual located at the owner controlled area boundary. The analysis demonstrated the dose to this individual would meet the requirements of 10 CFR 72.104. (Section 3.2.e)

- The licensee was maintaining 10 CFR Part 72 records in their quality related records system. (Section 3.2.f)

Review of 10 CFR 72.48 Evaluations (60857)

- Safety screenings had been performed in accordance with the licensee's procedures and 10 CFR 72.48 requirements. All screenings reviewed were determined to be adequately evaluated. One 10 CFR 72.48 evaluation identified three areas (fire hazards, tornado missiles, and transfer cask drop scenario) where implementation of the UMAX storage system at the SONGS site was identified to be different than the descriptions provided in the HI-STORM FW and UMAX Final Safety Analysis Reports. All three changes were evaluated by the licensee through the site's 10 CFR 72.48 process to demonstrate the evaluations continued to meet the system's original design basis acceptance criteria listed in the HI-STORM FW and UMAX Final Safety Analysis Reports. An Unresolved Item was opened to track the NRC's review of the methodology

utilized in the evaluation for the transfer cask drop within the spent fuel pool and determine if the change could be performed through the 10 CFR 72.48 process. (Section 4.2.a)

Report Details

Summary of Facility Status

The SONGS ISFSI consists of two ISFSI designs located adjacent to each other. The Transnuclear, (TN) Inc. Nuclear Horizontal Modular Storage (NUHOMS) ISFSI contained 51 loaded concrete advanced horizontal storage modules (AHSMs) which housed stainless steel dry shielded canisters (DSCs). Spent fuel from all three reactors were stored at the NUHOMS ISFSI in 50 of the canisters. Greater-than-Class-C (GTCC) waste from the Unit 1 reactor decommissioning project was stored in one canister. There were a total of 63 AHSMs on the NUHOMS ISFSI pad. The twelve empty AHSMs will be available for storage of additional GTCC waste. The NUHOMS ISFSI pad consisted of two adjacent pad areas designed to hold the AHSMs. The pads were both 293 feet in length. The first pad area was 43 feet 6 inches wide and held 31 canisters. The second pad area was 60 feet 6 inches wide and was designed to hold 62 AHSM in a double row, positioned back to back. The 63 AHSMs currently on the TN ISFSI pads were designed for the 24PT1-DSC (Unit 1 fuel) and 24PT4-DSC (Unit 2/3 fuel) canisters, which hold a maximum of 24 spent fuel assemblies. The 24PT1-DSCs were loaded and maintained under Amendment 0 of Certificate of Compliance (CoC) 72-1029 and the 24PT4-DSCs were loaded and maintained under Amendment 1 of the CoC 72-1029. Both systems were being maintained under Final Safety Analysis Report (FSAR) Revision 5.

The Holtec UMAX ISFSI portion was designed to hold 75 multi-purpose canisters (MPCs). The UMAX ISFSI is 231 feet long and 102 feet wide. However, its dimensions are not rectangular. The ISFSI is wider on its northern end than on its southern end. The support foundation pad was constructed below grade at the 8.5’ Mean Lower Low Water (MLLW) elevation. The top of the ISFSI top pad was located at the 31.5’ MLLW elevation. Approximately half of the UMAX ISFSI was located below grade while the other half had excavated common fill that sloped up to the top of the ISFSI top pad. The licensee has begun loading MPC-37s containing 37 pressurized water reactor fuel assemblies in accordance with UMAX CoC No. 72-1040 and Technical Specifications, Amendment 2, the HI-STORM UMAX FSAR, Revision 4, and the HI-STORM FW FSAR, Revision 5. The licensee plans to remove all the remaining fuel from the Units 2 and 3 spent fuel pools to the UMAX ISFSI.

1 Preoperational Testing of an ISFSI at Operating Plants (60854)

1.1 Inspection Scope

The NRC inspectors reviewed by direct observation and independent evaluation that the licensee has developed, implemented, demonstrated, and evaluated preoperational testing activities to safely load spent fuel into a dry cask storage system and transfer the loaded canister to the ISFSI. The inspections verified the licensee fulfilled all appropriate testing acceptance criteria and implemented all required changes to the appropriate plant programs and procedures to support ISFSI operations.

1.2 Observations and Findings

a. Canister Drying

The licensee utilized forced helium dehydration (FHD) to achieve the dryness levels required by Technical Specification Appendix A, Table 3-1. The operation of the system

was described in procedure HPP-2464-300 “MPC Sealing at SONGS,” Revision 0. The NRC inspectors verified that the licensee met the technical specifications required limits for dryness during the loading of the first canister in the January 2018 inspection. Helium meeting the Technical Specification, Appendix A, Table 3-1 requirement for a purity of 99.995 percent or greater was verified to be utilized during dry run demonstrations and first loading operations associated with MPC blowdown, drying, and backfill operations. Helium backfill pressure requirements were incorporated into licensee procedure HPP-2464-300. The NRC inspectors observed that the required backfill pressure was met during the loading of the first canister.

b. Crane Design and Loading Operations Seismic Analysis

The licensee utilized 125-ton Ederer’s Extra Safety and Monitoring (X-SAM) single-failure-proof cranes in each of their Unit 2 and Unit 3 spent fuel buildings to transfer the MPC and transfer cask (HI-TRAC VW) out of the spent fuel pool to the cask washdown area and then onto the low-profile transporter (HI-PORT). The NRC had reviewed the safety features of the X-SAM crane and issued a Safety Evaluation Report on January 2, 1980, related to Ederer’s Generic Licensing Topical Report EDR-I(P), “Ederer’s Nuclear Safety Related Extra Safety and Monitoring (X-SAM) Cranes,” Revision 1 and on August 26, 1983, related to Revision 3. In the 1980 letter, the NRC stated that the design features presented in the topical report for the Ederer X-SAM crane were acceptable for assuring that a single failure would not result in the loss of capability to safely retain a critical load. In the 1983 letter, the NRC Safety Evaluation Report discussed the features of the wire rope used for the X-SAM crane and noted the safety criteria for the wire rope was met and was found acceptable to the NRC.

The fuel building overhead crane used a dual rope reeving system with individual attaching points and a load balancing system to hold and transfer the critical load without excessive shock in case of failure of one of the rope systems. The X-SAM crane is equipped with an energy absorbing torque limiter (EATL) which allows the hoist to safely withstand two blocking, overloading, or load hang-up, and still retain the load even if the drive motor is de-energized. Not only are the loads controlled following a two-blocking, load hang-up, etc., but the hoist’s components are also protected, throughout their life, from being overstressed by these incidents. To provide this protection, the EATL directly converts the hoists high speed kinetic energy to heat during an overloading incident. The crane also utilized a system of upper travel limit switches that were designed to shut the crane down before a two-blocking event could occur.

The hoist drum was provided with the structural and mechanical safety devices to limit its drop during a shaft or bearing failure. The devices would also prevent disengaging from the holding brake. Ederer Topical Report EDR-I (P)-A, Section III.B.1.b, stated “The emergency drum brake system provides an independent means for reliably and safely stopping and holding the load following a failure in the hoist machinery.” Hoist machinery failures included shaft or bearing failures. The crane was designed to retain control of and hold loads during seismic events. The bridge and trolley were designed to remain in place on their respective runways with their wheels prevented from leaving the tracks during a seismic event.

All of the Licensee’s 10 CFR Part 72 seismic evaluations, for use of the UMAX system, were reviewed by NRC Division of Spent Fuel Management (DSFM) during the

inspection period. This review included seismic loading analysis for cranes, as well as the seismic stability analysis of the transfer operations of the MPC to the ISFSI pad. The seismic stability during transfer operations included the HI-TRAC VW transfer cask (loaded with an MPC) in the spent fuel building during decontamination and closure operations, on the HI-PORT, on the vertical cask transporter (VCT), and during transfer of the MPC to the UMAX storage system ISFSI.

The rated load and seismic analysis was conducted using GT-STRUDL to analyze a three-dimensional model to create the mass and stiffness properties of the crane components using line elements and lumped masses. The response spectrum method from American Society of Mechanical Engineers (ASME) NOG-1, “Rules for Construction of Overhead and Gantry Cranes,” was used in the analysis of the seismic loads. The load combinations applied to the model were consistent with those of Crane Manufacturers Association of America, Inc. (CMAA)-70 “Specification for Top Running Bridge and Gantry Type Multiple Girder Electric Overhead Traveling Cranes,” (2000) which included Operational Basis Earthquake (OBE) and Design Basis Earthquake (DBE) loads as well as the 125-ton live load, which is the rated capacity of the crane. The three orthogonal components of the earthquake motion were combined using the square root sum of squares of the structural response and combined with the static load cases. A two percent critical damping was used for OBE case and a four percent critical damping was used for the DBE case. Hand calculations and the finite element software ANSYS were used to analyze the forces on the individual components to determine their acceptability. The codes, standards and regulations used for the analysis and acceptance criteria included ASME B30.2 (1996); CMAA-70; ASME NOG-1 (2000); American Society of Civil Engineers 4-86, “Seismic Analysis of Safety-Related Nuclear Structures” (1986); NUGREG-0554, “Single Failure Proof Cranes for Nuclear Power Plants,” (1976); American Institute of Steel Construction (AISC) Manual of Steel Construction, 9th edition; American Welding Society (AWS) D1.1, “Structural Welding – Steel;” AWS D14.1, “Specification for Welding of Industrial and Mill Cranes and other Material Handling Equipment;” and American National Standards Institute (ANSI) N14.6, “Special Lifting Devices for Shipping Containers Weighing 10,000 Pounds or More,” (1993).

As part of the analyses, members classified as non-compact according to the AISC, were checked for local buckling. Several upgrades were completed to satisfy the seismic qualification of the 125-ton crane, including a 12-wheel trolley option in lieu of the 4-wheel trolley. Other specific upgrades included: replacing bolts in connection between the girder and the truck, adding fillet welds between the lower connection plate and the bottom of the bridge truck, adding a shim plate to the inside face of the box girder top flange (the shim provided a contact surface for the X-SAM trolley uplift seismic restraints), adding longitudinal stiffeners below the top flange, and adding vertical/transverse stiffeners to limit the web panel size to 48-inches to satisfy CMAA-70 and ASME NOG-1 web buckling requirements.

Based on the review of the design documents and calculations, the DSFM staff concluded that there was reasonable assurance that the cranes were structurally adequate to withstand the earthquake loads during fuel loading operations.

The HI-TRAC VW loaded with the MPC containing spent nuclear fuel was analyzed using a 1.20g zero period acceleration at the floor level of the cask wash down area. The HI-TRAC VW was prevented from tipping over by restraints at two levels that connect to the wall of the cask wash down area. The restraints consist of two slings that connect to the wall mounted attachments and wrapped around the cask in a crisscross fashion to prevent the cask from tipping over. The analysis included a concrete wall evaluation, a base plate and anchor bolt evaluation, and a transfer cask stop evaluation.

The concrete wall evaluation demonstrated that the wall had sufficient strength to withstand the added bending and shear forces caused by the seismic loads on the cask, to include impact with the wall. In addition, should the concrete cask impact the wall, the wall had sufficient thickness to prevent penetration or perforation, and sufficient strength to resist the punching shear that results from compression on the steel tubes that make up the cask stop.

The analysis of the seismic restraint anchor assembly demonstrated that the base plate, stiffener plates and associated welds, and anchor bolts had sufficient strength to withstand the seismic loads due to restraining the cask.

The transfer cask stop consisted of a steel tubes connected together with welded gusset plates. The analysis of the stop assembly determined that the steel tubes, gusset plates and associated welds were structurally adequate to resist the compressive, bending, and shear forces due to the seismic load. Additionally, the force generated from the seismic load was within the load capacity of the seismic restraints and shackle.

Based on a review of the design documents and calculations, the DSFM staff concluded that there was reasonable assurance that the seismic restraint system as well as the concrete wall to which it was attached, had adequate strength to maintain the HI-TRAC VW transfer cask, loaded with an MPC and spent nuclear fuel, stable in the cask washdown area under the DBE.

The HI-PORT, loaded with the HI-TRAC VW and MPC, during transit on the haul path at SONGS was analyzed for stability (tip-over and sliding) during a design basis seismic event. The HI-PORT was comprised of two trailers with a drop deck between them. The HI-TRAC VW bottom flange was bolted to a seismic restraining ring which was bolted to the drop deck of the HI-PORT.

Five time history sets were used to perform the stability analysis which was simulated with the computer code LS-DYNA. The mean values of peak axial and shear loads on the individual bolts were obtained from the dynamic analysis, as were the mean bending and shear loads in the trailers and drop-deck, and the mean loads at the connections between the trailers and the drop-deck. These loads were compared against the structural capacities of the respective components. All load bearing components were shown to have safety factors greater than 1.0 (structural capacity was greater than structural demand). The maximum rocking angle in the lateral direction was 0.035 degrees and the maximum sliding distance of the HI-PORT was 10.38 inches. Using a factor of safety of three, a minimum clearance of 32 inches to the outer edge of safety related structures was established and implemented in the licensee's transportation procedures. In addition, the HI-PORT was restricted to 3.1 miles per hour.

Based on a review of the design documents and calculations, the DSFM staff concluded that there was reasonable assurance that the HI-PORT, loaded with the HI-TRAC VW transportation cask, would not tip over, and that the HI-TRAC VW would remain attached to the HI-PORT during a DBE. Additionally, with the imposed transport limitations (distance and speed), the HI-PORT would not impact safety related structures while in transit during a potential DBE.

The seismic response of the VCT carrying the HI-TRAC VW was analyzed on the haul path, the transfer slab, the ISFSI ramp, the approach slab, and the ISFSI pad during the bounding DBE. The design basis response spectra and corresponding time histories at grade level were used in the stability evaluation to ensure the VCT did not tip over and remained on the respective path, transfer/approach slab, and ISFSI pad.

The ISFSI ramp was assumed to have a grade of seven percent. Based on Licensee UMAX design drawings, the maximum grade of six percent existed on the ISFSI ramp. Additionally, the VCT was assumed to tip in the lateral direction (shortest footprint dimension), which would require the VCT, loaded with a HI-TRAC VW, to travel across the path instead of up or down the path. The site specific zero period acceleration for SONGS was 0.67g horizontal and 0.45g vertical. The amplification from the HI-STORM UMAX soil structure interaction (SSI) analysis was 1.1, 1.0, and 1.08 in the E-W, N-S, and vertical directions for the top of the ISFSI pad. The zero period acceleration was amplified by 15 percent for the analysis on the ISFSI pad, approach slab, and ramp.

The center of gravity of the VCT loaded with the HI-TRAC VW was based on a maximum lift height of 11 inches on the haul path and 51 inches on the ISFSI pad. These lift height distances were controlled by the licensee’s transfer operation procedures.

Upon review of the sliding analysis, it was determined that the VCT will slide under the bounding DBE. A minimum distance of 47 inches from the edge of the ISFSI ramp, approach slab, and ISFSI pad was recommended to ensure the VCT would not slide off of the structures. This limit was based on a safety factor of greater than 1.0. The licensee’s transportation procedure contained the required standoff distance and a white line was painted around the edge of the ISFSI ramp, approach slab, and ISFSI pad to ensure workers would abide by the limitations from the evaluation.

Based on a review of the design documents and calculations, the DSFM staff concluded that there was reasonable assurance that the VCT, loaded with the HI-TRAC VW transfer cask, would not tip over on the transfer slab, ISFSI ramp, approach slab, or the ISFSI pad as a result of the DBE. Additionally, with the imposed transport limitations, the staff had reasonable assurance that the VCT, loaded with the HI-TRAC VW, would not slide off of the ISFSI ramp, approach slab, or the ISFSI pad as a result of the DBE.

The stack-up evolutions at the UMAX ISFSI pad consisted of the HI-TRAC VW transfer cask bolted to the Mating Device (MD), the MD bolted to the Mating Device Adapter (MDA), and the MDA bolted to the HI-STORM UMAX Cavity Enclosure Container (CEC). An evaluation was performed to determine the structural adequacy of

the HI-TRAC VW-to-MD, MD-to-MDA, and MDA-to-CEC connections as well as the ISFSI pad bearing capacity under the DBE.

A finite element model of the HI-TRAC VW, MD, and MDA on top of the ISFSI pad was built in LS-DYNA to determine the loading on the bolts, welds, and components, as well as the ISFSI pad. Hand calculations were then used to determine the structural adequacy of the connections and components in accordance with ASME Boiler and Pressure Vessel Code (BPVC), Section III, Division I, Subsection NF, and the structural adequacy of the ISFSI pad in accordance with American Concrete Institution (ACI) 318-05. A scale factor of 20 percent was applied to the at-grade DBE basis earthquake time history set in all directions to account for amplification at the top of the pad.

The peak axial and shear loads on the bolts that connected the HI-TRAC VW, MD, MDA and CEC were all less than the maximum allowable load for the bolts. The bolt interaction ratio (used to evaluate the combination of axial and shear forces on the bolts) were less than one, indicating the bolts were adequate under the combined axial and shear forces. Additionally, an analysis of the shear strength of the threads determined that the engagement lengths of the bolts were adequate for the connections.

The plate stresses in the MD were taken directly from the LS-DYNA analysis and compared with the allowable stress for that material. Components and welds that were not explicitly modeled were evaluated using bounding loads obtained from the analysis. All load bearing components and welds were determined to have safety factors greater than 1.0, meaning the calculated stress was less than the allowable stress for that material.

The tensile loads at the MD-to-MDA and MDA-to-CEC bolted connections were used to evaluate the supporting components and welds within the MDA. All bearing components and welds were determined to have safety factors greater than 1.0.

Finally, the ISFSI pad concrete bearing capacity was evaluated using the total load along each side of the MDA that was extracted from the LS-DYNA analysis. The safety factors against bearing on the ISFSI pad concrete due to the loads between the MDA and the CEC cover plate during stack-up were determined to be greater than 1.0.

Based on a review of the design documents and calculations, the DSFM staff concluded that there was reasonable assurance that the stack-up of the HI-TRAC VW, MD, and MDA on the CEC had adequate strength to sustain the DBE on the ISFSI pad. Additionally, the staff concluded that the ISFSI pad concrete strength was sufficient to withstand the DBE during stack-up operations.

c. Crane Inspection and Operation

During the licensee's programs review, NRC inspectors reviewed SONGS crane maintenance program for the 125-ton single-failure-proof X-SAM cranes located in the Unit 2 and 3 spent fuel buildings. Frequent crane inspections were performed daily during use, on the X-SAM cranes as required by the ASME B30.2 code. The inspection criteria from the ASME B30.2 code was captured in the licensee's Procedure HPP-2464-010, "SONGS Cask Handling Crane Checkout and Operation,"

Revision 2. The NRC inspectors observed the licensee perform the daily inspection during dry run demonstrations and first canister loading operations.

The required annual testing of the overhead X-SAM cranes followed HPP-2464-009, “Maintenance and Inspection of Cranes,” Revision 1. The latest annual inspection was completed during the recent load testing of the cranes on November 11, 2017, for Unit 2 and October 2, 2017, for Unit 3. The licensee’s procedure contained all the required inspection criteria outlined in ASME B30.2 and ASME B30.10, “Hooks.” Additionally, all the crane’s safety devices were tested in accordance with the Ederer Topical Report, Revision 3. The safety devices tested included: overload sensing system, hydraulic load equalization system fluid level, EATL, emergency drum brake system, drive train continuity detector, and wire rope spooling monitor.

Crane operation requirements and crane operator qualification requirements from ASME B30.2 were reviewed during dry run demonstrations and the first loading operations by NRC inspectors. The NRC inspectors verified that the crane operators training and qualification program met the requirements of the ASME code. Documentation was provided that demonstrated the crane operators for the first loading operations were trained and qualified in accordance with the licensee’s program. The NRC inspectors observed the operators perform the required ASME code brake test prior lifting a load that approached the rate load. This was accomplished by raising the load a short distance and applying the brakes to ensure the load would not lower unexpectedly. In accordance with the site’s heavy load program and NUREG-0612, “Control of Heavy Loads and Critical Lifts,” lift heights, load paths, special provisions, temperature restrictions, and rigging diagrams were placed in the appropriate procedures for the transfer operations that were occurring.

d. Crane Load Testing

The maximum calculated weight of the HI-TRAC VW with a MPC loaded with spent fuel and water raised out of the spent fuel pool was described in Holtec Report No. HI-2156458, “Cask Handling Weights at SONGS,” Revision 3 as 246,537 pounds (123.3 tons). Both Units’ 125-ton X-SAM cranes had recently completed a static load tested to 125 percent the rated capacity followed by a dynamic performance load test at 100 percent of the rated capacity. The Unit 2 crane’s load testing was completed on November 20, 2017, and the Unit 3 crane’s load testing was completed on October 2, 2017. The dynamic testing included movement in all directions and verifying all limiting and safety control devices. Additionally, the licensee provided documentation that demonstrated that each of the 125-ton hooks had been statically load tested to 200 percent the rated capacity in accordance with ASME B30.10 in 2003 by Ederer Inc.

e. Dry Run Demonstrations

The Holtec CoC 72-1040 Condition #8 required that dry run training exercises of the loading, closure, handling, unloading, and transfer of the HI-STORM UMAX Canister Storage System shall be conducted by the licensee prior to the first use of the system to load spent fuel assemblies. The dry runs shall include, but are not limited to the following: (a) Moving the MPC and the transfer cask into the spent fuel pool or cask loading pool; (b) Preparation of the HI-STORM UMAX Canister Storage System for fuel loading; (c) Selection and verification of specific fuel assemblies to ensure type

conformance; (d) Loading specific assemblies and placing assemblies into the MPC (using a dummy fuel assembly), including appropriate independent verification; (e) Remote installation of the MPC lid and removal of the MPC and transfer cask from the spent fuel pool or cask loading pool; (f) MPC welding, nondestructive examination (NDE) inspections, pressure testing, draining, moisture removal (by vacuum drying or forced helium dehydration, as applicable), and helium backfilling (A mockup may be used for this dry-run exercise); (g) Transfer of the MPC from the transfer cask to the HI-STORM UMAX Vertical Ventilated Module (VVM); and (h) HI-STORM UMAX Canister Storage System unloading, including flooding MPC cavity and removing MPC lid welds (A mockup may be used for these dry-run exercises).

On June 26-30, 2017, NRC inspectors observed SONGS perform dry run demonstrations listed in Condition #8 (f) and (h): MPC welding, NDE inspections, and removing MPC lid welds. The licensee utilized Holtec's welding vendor PCI Energy Services (PCI) to perform the welding on a mock-up canister. The welding demonstration included MPC lid to shell welding, welding of the vent and drain cover plates, welding of the plug on the cover plates, welding of the canister closure ring, and demonstration of the in-line hydrogen monitoring system. The visual NDE examinations and the liquid dye penetrant examinations were performed on all the welds. Additionally, helium leak testing of the vent and drain port covers was performed during the dry run by Leak Test Services (LTS). The licensee successfully demonstrated all required welding and the NDE examinations.

The removal of the canister lid welds was demonstrated by providing the NRC with a videotape of a welded MPC-37 lid being removed. The DSFM has accepted that if the cutting evolution had been successfully completed on the same model of MPC canister at one site, another general licensee can take credit for the demonstration, as long as the same equipment and procedures would be utilized. The demonstration to remove the welds from a MPC-37 canister was performed July 16-18, 2015, at the Holtec Manufacturing Division located in Turtle Creek, PA. Inspectors from NRC's DSFM observed the cutting dry run at the Holtec facility. The cutting activities included boring through the cover plate and the MPC vent/drain port covers. The lid cutting machine was then utilized to cut through the cover plate and the MPC lid-to-shell weld. During the cutting evolution, Holtec personnel purged the area under the lid with argon while monitoring for hydrogen as required by the FSAR. All cutting demonstrations were successful, and the MPC lid was removed from the shell. This inspection was documented in an NRC Inspection Report (ADAMS Accession No. ML15303A348). The procedures and arrangements to use the same cutting system had been adopted into the SONGS ISFSI program.

On August 1-3, 2017, NRC inspectors observed SONGS complete dry run demonstrations of Condition #8 (f) and (h). The specific operations included: pressure testing, draining, moisture removal (by forced helium dehydration), helium backfilling and the unloading portion of flooding the MPC cavity. The fluid operations demonstration included observing the licensee's implementation of their radiation protection and foreign material exclusion programs. All demonstrations were successfully performed on a mock-up canister.

On September 25-28, 2017, NRC inspectors observed SONGS complete dry run demonstrations of Condition #8 (b), (g), and (h). The specific operations included:

preparation of the UMAX for canister loading, transfer of the MPC/transfer cask from the spent fuel pool building to the UMAX ISFSI, downloading the MPC into the VVM, and unloading portions that included removing the MPC from the VVM and returning the MPC/transfer cask to the spent fuel building. The heavy loads demonstration included preparing the UMAX for the canister by installing the mating device, use of the HI-PORT and the VCT to move the canister from the spent fuel pool building to the UMAX ISFSI and back. All demonstrations were completed with a mock-up canister that was filled with concrete to simulate the weight of the MPC loaded with spent fuel. The licensee successfully completed all required movements associated with the required demonstration.

On October 9-13, 2017, during the programs review, the inspectors reviewed the licensee's fuel selection and verification procedure completing dry run demonstration Condition #8 (c). Additional information related to the fuel selection is contained in Section 1.2.f of this report. Additionally, a physical walk-through of the selection and verification process associated with the licensee's program was demonstrated during the final dry run when the licensee performed fuel loading operations of a dummy fuel assembly into several positions in the canister basket on December 4-7, 2017. The licensee successfully implemented an adequate process to select fuel and to verify the assemblies loaded.

On December 4-7, 2017, the NRC inspectors observed SONGS complete dry run demonstrations of Condition #8 (a), (c), (d), and (e). The specific operations included: moving the MPC and the transfer cask into the spent fuel pool, a walk-through of the independent verification process for fuel loading, loading a dummy fuel assembly into a number of positions in the MPC, remote installation of the MPC lid, and removal of the MPC and transfer cask from the spent fuel pool. These operations were completed in the Unit 3 spent fuel building using the licensee's 125-ton overhead cask handling crane and the Unit 3 bridge crane that moves fuel assemblies within the pool. This demonstration completed all the required dry run demonstrations from the CoC. The licensee successfully completed the above listed operations and demonstrated that the procedures, programs, and training related to the dry cask storage operations for the Holtec HI-STORM UMAX system had been successfully integrated into their site operations.

f. Fuel Selection/Verification

Dry cask storage planning for the SONGS UMAX ISFSI included removing all fuel contents from the Unit 2 and 3 spent fuel pools (SFPs) to support decommissioning activities at the formerly operational nuclear plant. The items to be placed into the UMAX ISFSI included 2,668 spent fuel assemblies and associated hardware, Rod Storage Baskets, and other fuel associated debris from the two SFPs. The NRC inspectors reviewed Holtec Report HI-2167416, "Loading Plans for SONGS ISFSI Expansion," Revision 6. All of the SFP contents to be stored in the SONGS ISFSI met the HI-STORM UMAX CoC 72-1040, Appendix B requirements for storage of spent fuel assemblies, damaged fuel assemblies, and other associated fuel related items. The spent fuel planned for storage in the SONGS UMAX ISFSI also met the loading requirements of the proposed Holtec HI-STAR 190 transportable cask.

The licensee performed a full characterization of the spent fuel contents of their Unit 2 and 3 SFPs. The fuel assemblies selected for storage met all of the Holtec CoC 72-1040 requirements, including length, width, weight, cooling time, fuel utilization (burn-up), cladding types, decay heat, and fuel initial enrichment. The majority of the contents to be loaded into the Holtec UMAX ISFSI were intact spent fuel assemblies. There were, however, a number of a fuel assemblies that met the Holtec UMAX CoC Appendix B definition of damaged fuel assemblies. The items identified as damaged fuel or fuel debris can be stored in the UMAX ISFSI but can only be loaded into twelve peripheral locations of the MPC-37 canister in damaged fuel containers. Approximately 28 MPC-37s with damaged fuel containers will be loaded into the SONGS UMAX ISFSI.

In the event of an MPC misloading (violation of CoC 72-1040, Appendix B, Section 2.1), SONGS Procedure SO123-0-A7, "Notification and Reporting of Significant Events," Revision 44, required that SONGS notify the NRC Operations Center within 24 hours after the licensee or other entity discovers the violation.

Procedure HPP-2464-200, "MPC Loading at SONGS," Revision 0 included steps that address the requirements of Holtec CoC 72-1040, Appendix A, including meeting the proper boron concentrations for loading the intact and damaged spent fuel assemblies at SONGS. The procedure included steps for independent post loading verification of fuel assemblies by SONGS Reactor Engineering personnel by video. The post loading verification is required by the HI-STORM FW FSAR, Section 9.2.3.3. Site procedures provided provisions for controlling and tracking the stored spent fuel records in accordance with 10 CFR 72.72 and 10 CFR 72.174. In accordance with the requirements of 10 CFR Part 74, SONGS Procedure SO123-X-1.7, "Special Nuclear Material Accountability," Revision 22 controlled tracking spent fuel and special nuclear material.

g. Heavy Loads

The licensee utilized two VCTs to lift the loaded HI-TRAC VW with MPC from the HI-PORT to the UMAX ISFSI pad for long term storage. The VCT was classified as an Important to Safety (ITS) component since the device provided the function of a crane to download the MPC from the HI-TRAC VW into the CEC. Each VCT was factory tested, statically to 125 percent and dynamically to 100 percent of the rated load. The VCTs were rated to 207.5 tons, in order to accommodate users that utilize the same VCT to carry a loaded HI-STORM FW overpack that weighs considerably more than a loaded HI-TRAC VW (118.5 tons). One VCT was tested on April 9, 2015, the other on April 7, 2016. All the weights utilized were verified to be slightly over the 125 percent and 100 percent weight requirements. During the dynamic load test, each VCT was traveled in all directions while testing the systems' safety devices.

The VCT's MPC downloader system was statically tested to 150 percent and dynamically to 100 percent of the rated load on the same dates as the VCT load testing described above. The MPC downloader system was rated to 128 tons. The weight of an MPC loaded with spent fuel and backfilled with helium weighed approximately 49 tons. After the testing of each downloader system, all accessible load bearing welds for the VCT that were designated as ITS, were subjected to visual and magnetic particle testing.

Technical Specification 5.2.c.2 required the VCTs to be inspected and maintained in accordance with NUREG-0612. Based on Holtec guidance, the licensee inspected the transporter in accordance with applicable sections of ASME B30.2 to meet the requirement. The daily inspection guidance was provided in HPP-2464-400, "MPC Transfer at SONGS," Attachment 8.8, "VCT Frequent Use Inspection Checklist." The annual inspection guidance was provided in HPP-2464-720, "Inspection and Maintenance for Vertical Cask Transporter," Revision 2 and was last completed on December 15, 2017 for each VCT. The inspection procedure met the applicable requirements of the ASME code.

The NRC inspectors verified that the transportation procedures associated with the VCT movements contained lift heights, load paths, special provisions, temperature restrictions, and rigging diagrams for all heavy lifts in accordance with the site's heavy load program and NUREG-0612 requirements.

h. Nondestructive Examination (NDE)

The NDE program adopted by SONGS to perform NDE inspections on the MPC welds was reviewed by the NRC inspectors to ensure the program and implementing procedures met the applicable ASME codes required by the UMAX FSAR. The NDE inspections of welds were performed by PCI's personnel. The helium leak testing was performed by LTS. During the welding dry run inspection on June 26-30, 2017, NRC inspectors reviewed the qualification requirements for the Level II or Level III inspectors for each program, the procedures utilized for each type of inspection, the work process, and the qualification of materials utilized in the inspections to verify the ASME/ANSI code requirements and technical specifications of license were properly incorporated in to licensee's program.

The helium leak testing was performed in accordance with ANSI N14.5, "Leak Tests on Packages for Shipment for Radioactive Materials," Revision 1997, to the established leak tight criteria of a leakage less than $2x10^{-7}$ atmosphere cubic centimeters per second (atm*cc/sec) as required by CoC 72-1040 Technical Specification, Appendix A Surveillance Requirement 3.1.1.3. The leak testing was performed in accordance with Procedure MSLT-MPC-Holtec, "Helium Mass Spectrometer Leak Test Procedure for MPC," Revision 3665-00. The process utilized a helium leak rate detector with a sensitivity level well below the technical specification leak rate criteria. Additionally, a calibration standard traceable to the National Institute of Standards and Technology was utilized to calibrate the helium leak rate detector prior to use. Four LTS Level III inspectors' certificates of qualification were reviewed to verify their certifications met American Society for Nondestructive Testing Inc. (SNT-TC-1A), "Recommended Practices for Qualification and Certification of NDE testing Personnel," Revision 1992 criteria and were current for the dates of the dry run and first loading inspection. During the first loading inspection, the licensee successfully performed the leak testing of the first MPC and results were below the required helium leak rate limit.

The NDE visual testing of the MPC canister welds was performed in accordance with Procedure GQP-9.6, "Visual Examination of Welds," Revision 16. The NRC inspectors verified the procedure contained the required acceptance criteria listed in ASME BPVC, Section III, "Rules for Constructions of Nuclear Facility Components," Article NF-5360,

Revision 1995. The procedure's qualification record demonstrated that the examination process was adequate to identify the required standard reference indications.

The NDE liquid penetrant testing of the MPC canister welds was performed in accordance with Procedure GQP-9.2, "High Temperature Liquid Penetrant Examination and Acceptance Standards for Welds, Base Materials, and Cladding," Revision 9. The NRC inspectors verified the procedure contained the minimum elements from ASME BPVC Section V, "Nondestructive Examination," Article 6, T-621, and the acceptance criteria listed in ASME Section III, NB-5352. The procedure's qualification record was reviewed to verify the process was capable of detecting the required indications. Certified mill test reports with chemical analysis for the materials used in the high temperature liquid penetrant examinations (cleaner solvent, developer, and dye penetrant) met ASME Section V, Article 6, T-641 requirements. All cleaning, developing, and final interpretation time limits, based on the temperature of the component, were specified in the procedure and adhered to by the NDE personnel. The liquid penetrant examination was required by the procedure to be performed on the root pass weld, prior to any intermediate weld exceeding 3/8", and the final weld in accordance with CoC 72-1040 Appendix B Table 3-1 criteria. The NDE personnel complied with ASME code requirements regarding surface preparation and avoiding excess penetrant removal. Two PCI Level II inspectors certifications of qualification were reviewed to verify their training was current and in accordance with the SNT-TC-1A qualification requirements for visual and liquid dye penetrant examinations. During the first loading inspection, the licensee successfully performed the NDE examinations on first MPC with no indications identified.

i. <u>Pressure Testing</u>

The Holtec HI-STORM UMAX FSAR states that the Holtec MPCs placed into the UMAX VVM for storage are pressure tested in accordance with Section III, Subsection NB-6000 of the ASME BPVC to meet structural requirements and to verify the confinement function of the UMAX dry fuel storage system. The UMAX FSAR established the MPC pressure testing requirements by making direct reference to the pressure testing requirements listed in the HI-STORM FW FSAR. Both HI-STORM FW and HI-STORM UMAX dry fuel storage systems utilize the MPC-37. In addition, the Holtec HI-TRAC VW water jacket was required to be hydrostatically pressure tested per the applicable ASME code after being manufactured and the test results documented.

Holtec HI-STORM FW FSAR, Section 10.1.2.2.2, "MPC Confinement Boundary," required that either a hydrostatic test to 125 percent of the design pressure or a pneumatic pressure test to 120 percent of the design pressure take place in accordance with the requirements of the 2007 ASME Code when field welding of the MPC lid-to-shell weld was completed. The design pressure of the MPC-37 canister is 100 psig.

The NRC inspectors reviewed Procedure HPP-2464-300, "MPC Sealing at SONGS," Revision 0, and found that the procedure described the hydrostatic testing of the MPC lid-to-shell weld, including holding the pressure between 125.5 to 129.5 psig for 10 minutes, and specified that the pressure be maintained. During the pressure test, the weld area was to be inspected for water leakage. After the test was completed, the canister was allowed to depressurize and a liquid dye penetrant test of the weld area was required. The steps of the procedure were aligned with the requirements of ASME code.

The NRC inspectors observed SONGS successfully perform the hydrostatic testing requirements of a mock-up MPC-37 canisters during the fluid operations dry run demonstration on August 1-3, 2017, and during the NRC inspection of loading activities for the first MPC-37 processed during the loading campaign on January 25, 2018. The hydrostatic test and the post visual and liquid penetrant examinations were performed satisfactorily on both occasions in accordance with ASME code requirements.

Procedure HPP-2464-300 controlled pressure gauge calibrations in accordance with ASME Code, Section III, Article NB-6413 to not exceed two weeks. The NRC inspectors verified that the pressure gauges used for the hydrostatic testing of the MPC had been calibrated within an acceptable date range during the first loading inspection.

j. Special Lifting Devices and Slings

The special lifting devices utilized for the UMAX loading operations were reviewed by the NRC inspectors to verify compliance with ANSI N14.6 requirements. The list of special lifting devices included: MPC lift cleats, HI-TRAC lift lugs, HI-TRAC lift links, lift yoke, and lift yoke extension. Component purchase specifications or structural evaluations of selected devices were reviewed to verify the material used for fabrication met the six times yield strength and ten times ultimate strength in accordance with ANSI requirements. Dual path components were required to be capable of lifting three times the combined weight of the shipping container plus the weight of the intervening components of the special lifting device, without generating a combined shear stress or maximum tensile stress at any point in the device in excess of the corresponding minimum tensile yield strength of the material of construction. The devices were also required to be capable of lifting five times the weight without exceeding the ultimate tensile strength of the materials.

The required load testing documentation was provided for each special lifting device to verify the devices underwent 300 percent load testing at the manufacturer's facility. The test loads were held for ten minutes and then a visual, dimensional, and NDE inspection were conducted on the components. No NDE indications or issues were identified during the post load testing of the devices reviewed.

Annual inspection of the special lifting devices was established in the licensee's programs. Procedure HSP-355 "Annual Recertification of Special Lifting Devices," Revision 3, covered the annual inspection requirements for the MPC lift cleats, HI-TRAC lift lugs, HI-TRAC lift links, lift yoke, and the Holtec lift yoke extension. Procedure HPP-2464-030 "Testing and Inspection of Trans Nuclear Dry Fuel Storage Special Lifting Devices at SONGS," Revision 1, provided the instructions to perform the annual testing of the TN equipment. In accordance with ANSI requirements, the procedures required either a load test with a visual and dimensional test or a nondestructive test of the critical areas with a visual and dimensional test if the load test was omitted.

k. Storage Operations

The licensee had established procedures and work orders to perform the required daily vent or air temperature monitoring surveillances required by the technical specifications, monthly vent inspections for damage, and monthly/annual/five year inspections of the ISFSI and VVMs per FSAR requirements. The daily vent or temperature monitoring inspections was implemented in licensee Procedure S023-3-2.37 "Advanced Horizontal Storage Modules/Vertical Ventilated Modules System," Revision 9 in accordance with CoC 72-1040, Appendix A, Technical Specification 3.1.2. The monthly vent inspection for damage was implemented in licensee Work Order Task Sheet 0917-77051-3 "HI-STORM UMAX ISFSI VVM Vent Screens," in accordance with UMAX FSAR Table 10.4.1 requirements. The monthly, annual, and five year inspections of UMAX ISFSI and VVMs was implemented in a number of work orders which met the requirements listed in UMAX FSAR Tables 10.4.1 and 10.4.2.

l. Welding

The NRC inspectors reviewed the licensee's MPC closure procedure to ensure that the lid-to-shell weld, closure ring weld, and vent and drain cover welds met the requirements of CoC 72-1040, Appendix B, such that all applicable welds were subjected to liquid dye penetrant examination and helium leak testing, when applicable, and combustible gas monitoring was in place during the lid-to-shell welding. As required by CoC 72-1040 Condition 8.f (see Section 1.2.e, above), the licensee successfully demonstrated that their welding processes during the welding dry run demonstration on June 26-30, 2017. The NRC inspectors also verified that the CoC 72-1040, Appendix B requirements were satisfied during the processing of the first MPC-37 for SONGS' UMAX loading campaign.

During the welding dry run, the NRC inspectors verified that all of the applicable requirements of ASME BPVC Sections -II, -III, and -IX were being followed for welding materials, procedure qualification, and welding performance in the field. In specific, the NRC inspectors verified through procedure and document review that the appropriate weld qualification records were in place and that certain welding processes, such as tack welding, gas tungsten arc welding, and weld repairs, followed the appropriate guidance.

The NRC inspectors verified by records review that weld filler materials and electrodes met the minimum applicable requirements of ASME BPVC, Sections -II and -III, including delta ferrite content. The NRC inspectors also verified by procedure review and field verification that the licensee had procedures in place to direct the specification, control, and storage of purchased weld materials in accordance with 10 CFR 72.154.

The licensee had procedures in place to direct all welding activities, including weld repairs. The training and qualification records for the welders were provided for inspection. The welders performing the MPC closure operations during the dry runs and for the loading of the first MPC-37 met all of the required training and were qualified to perform all of the welds applicable to MPC-37 closure operations.

1.3 Conclusions

The FHD dryness limits, helium purity, and helium backfill requirements established in Technical Specification Appendix A Table 3-1 had been incorporated into the licensee's

procedures. The licensee planned to use the FHD system for drying all canisters loaded at the site. Operation of the FHD system and backfill to the required limits was demonstrated during the pre-operational dry run exercises and first loading activities.

The cask loading cranes used in the spent fuel handling buildings to lift the spent fuel canisters had been accepted by the NRC in 1980 as single failure proof cranes. The cranes were designed to retain control of and hold loads during a DBE at the SONGS site. Calculations were reviewed by NRC's DSFM that demonstrated that the forces from a seismic event in the upward and horizontal directions would not exceed the strength of the crane's seismic restraints. Additional seismic evaluations were reviewed to ensure seismic stability during transfer operations. This review included the transfer cask (loaded with a canister) in the spent fuel building during decontamination and closure operations, on the low profile transporter, on the vertical cask transporter, and during transfer of the MPC into the UMAX ISFSI. Based on the review of the design documents and calculations, the Division of Spent Fuel Management's staff concluded that there was reasonable assurance that the cranes and other handling/restraining equipment were structurally adequate to withstand DBE loads during fuel loading operations.

The 125-ton spent fuel building cranes were subjected to daily prior-to-use inspections that satisfied the requirements of ASME B30.2. On an annual basis the cranes were subjected to a more rigorous inspection that met the requirements of ASME B30.2 and the Ederer Generic Licensing Topical Report

The 125-ton spent fuel building cranes were properly load tested, as required by ASME B30.2, in the fall of 2017. The tests included a full performance test with 100 percent of the maximum critical load and a 125 percent static load test. The cranes' hooks were subjected to a 200 percent hook load test in 2003 by Ederer Inc.

The NRC inspectors observed the licensee successfully complete all the required pre-operational tests specified by License Condition #8 of the CoC. This included fuel assembly selection, welding, nondestructive testing, drying, helium backfilling, and the unloading of a sealed canister. A weighted canister was used to demonstrate heavy load activities inside the fuel handling building, transport between the fuel handling building and the ISFSI, and movement back into the fuel handling building for unloading purposes.

The licensee's fuel loading characterization plan met the HI-STORM UMAX CoC 72-1040, Appendix B limits for length, width, weight, irradiation cooling time, average burn-up, cladding, decay heat, and fuel enrichment. The licensee had established provisions for independent verification of the correct loading of spent fuel assemblies into the canister by use of video.

The licensee had incorporated the requirements related to the ISFSI project into the site heavy loads programs and procedures. Lift height limits, travel paths, and temperature restrictions during movement of the transfer cask had been incorporated into the licensee's procedures consistent with the requirements in the FSAR. The site's VCT were load tested and maintained in accordance with NUREG-0612 criteria.

The requirements for nondestructive testing of a spent fuel canister were incorporated into the licensee's procedures. The helium leak testing equipment used during the dry run demonstration and first loading was verified to meet the requirements listed in the technical specifications. The visual and liquid dye penetrant examination procedures implemented all the applicable requirements from ASME BPVC Section III, Section IV, and the FSAR regarding nondestructive examination of welds. A review of the nondestructive testing personnel's qualifications revealed they were properly qualified as a Level III or Level II examiners.

The requirements for canister hydrostatic testing had been incorporated into the licensee's procedures and were consistent with the requirements of ASME BPVC Section III Subsection NB, Article NB-6000. The hydrostatic testing sequence and criteria described in the FSAR had been incorporated into the licensee's procedures.

The licensee's special lifting device program complied with ANSI N14.6 criteria for stress design, annual inspections, and 300 percent proof loadings for the MPC lift cleats, HI-TRAC lift lugs, HI-TRAC lift links, lift yokes, and the lift yoke extensions.

The licensee had established procedures and work orders to perform the required daily monitoring surveillances required by the technical specifications, monthly vent inspections for damage, and monthly/annual/five year inspections of the ISFSI and VVM per FSAR requirements.

All welding procedures contained the required variables specified in ASME BPVC Section IX for gas tungsten arc welding. Requirements for hydrogen monitoring during welding of the inner cask lid had been incorporated into the procedures. The welder's performance qualification test records were reviewed and documented that the welders had met the qualification testing requirements for manual and machine welding of the canister lid. Weld qualification test coupons satisfactorily passed the required tests.

2 Operations of an ISFSI (60855)

2.1 Inspection Scope

The inspection included 24-hour coverage of the loading operations for the critical tasks associated with the licensee's first UMAX loading. Inspectors from NRC Region IV observed operations which included fuel loading, heavy lifts associated with the fuel building crane, welding and nondestructive testing of the canister lid-to-shell weld, hydrostatic pressure testing, forced helium dehydration, helium backfill, vent/drain port welding and nondestructive testing, helium leak testing, radiological surveys, and transport of the loaded HI-TRAC VW to the UMAX ISFSI pad. The inspectors reviewed selected procedures and records to verify ISFSI operations were in compliance with the Holtec CoC 72-1040 license technical specifications and Holtec FSARs.

2.2 Observations and Findings

a. Loading Operations

On January 22-31, 2018, NRC inspectors were onsite to observe the first canister loading operations. Inspectors observed all fuel assemblies loaded into the canister.

The fuel assemblies were inspected for damage prior to placement in the canister by use of an underwater camera. No damage was observed on any of the fuel assemblies loaded and the assemblies were free of foreign material. The canister's contents were reviewed to verify that the licensee was loading fuel in accordance with the technical specifications for approved contents. Documents reviewed included MPC loading maps and fuel assembly specific information such as identification, decay heat, cooling time, average U-235 enrichment, burn-up values, and other information. All fuel documents reviewed documented that SONGS had met the requirements listed in Appendix B of the CoC.

Observations of heavy lifts included placement of the MPC lid, removal of the HI TRAC VW with a loaded MPC from the spent fuel pool, placement of the HI-TRAC/MPC onto the HI-PORT, and lifting of the HI-TRAC/MPC from the HI-PORT to the VCT. The smooth operation of the 125-ton single failure proof crane and VCT was due, in part, to the licensee's extensive preventative maintenance effort on the lifting equipment. Numerous crane components had been replaced or upgraded to ensure successful completion of the upcoming continuous loading campaign. All lifting operations observed were performed in accordance with the site's heavy loads program.

Welding of the canister lid-to-shell weld began on January 24, 2018. The licensee utilized a calibrated in-line hydrogen monitor throughout the welding operations to ensure hydrogen levels were well below the lower explosive limit. Following the lid-to-shell welding, the required NDE (visual and dye penetrant testing) was performed to meet license requirements. No indications were identified during the NDE tests. Welding on the vent and drain port cover plates was completed after hydrostatic pressure testing, blowdown, FHD drying, and helium backfilling. The welds on the vent and drain port cover plates successfully passed all NDE examinations. After the vent/drain ports were helium leak tested, the closure ring was placed on the canister and properly welded.

The NRC inspectors observed the licensee successfully perform the hydrostatic pressure testing, blowdown, FHD drying, and helium backfill operations. The MPC was hydrostatically tested to the required pressure range, held for the required timeframe, and subsequently passed a second NDE exam. All water was then removed from the canister using the FHD and then successfully dried. The licensee met the time-to-boil time limit and had removed the water from the canister without having to initiate alternate cooling operations. The helium gas temperature exiting the freezer section of the dryer was below the required temperature and held for over 30 minutes in accordance with Technical Specification Appendix A Table 3-1, verifying the canister was adequately dried. The canister was then backfilled with helium of a purity greater than 99.995 percent, to the pressure range required in Technical Specification Appendix A Table 3-2.

Radiological coverage was provided throughout the loading campaign in accordance with the licensee's procedures. The radiation protection (RP) staff implemented adequate ALARA controls to minimize the overall collective dose during cask loading. The RP staff provided a sufficient amount of RP technician coverage during work activities, conducted detailed and comprehensive pre-job briefings on radiological conditions, effectively used portable radiation shielding, and effectively directed personnel to remain in low dosage areas when not actively working on the canister. The NRC inspectors observed the RP perform the required Technical Specification

Appendix A Section 5.3 surveys and verified the results were below the radiation and contamination limits specified.

During transportation operations to the ISFSI pad, NRC inspectors observed the licensee perform the required fire hazard walk-down of the haul path to ensure procedural requirements were met prior to transportation operations. The HI-PORT and VCT successfully transported the canister to the UMAX ISFSI without any malfunctions.

b. Design Control

During the first canister loading inspection on Monday January 22, 2018, the NRC inspector observed that the HI-TRAC VW transfer cask's seismic restraint system had been modified from its original design in order to be installed the Unit 2 spent fuel building. A 16 inch by 2 inch section of the back support plate for the seismic restraint system had been removed to allow the base plate to be installed around the existing sling restraints associated with the overall seismic restraint system. Additionally, the lift yoke extension had been non-structurally modified to be stored in the Unit 2 Spent Fuel Building. These design changes had been performed after the last NRC dry run inspection. The NRC inspector requested from SONGS the design change packages and applicable 10 CFR 50.59/72.48 reviews that were performed to ensure the newly modified ITS equipment would still be able to perform their safety function in accordance with the system's original design basis.

The licensee determined that the modification to both ITS components were processed through Holtec's field condition report (FCR) process under FCR-2464-LOA-065 for the seismic restraint base plate modification and under FCR-2464-LOA-041 for the lift yoke extension. The FCR-2464-LOA-065 for the seismic restraint base plate stated the system would continue to perform as designed, but the document did not contain sufficient technical analysis to justify the modification. The lift yoke extension FCR-2464-LOA-041 did contain the sufficient technical analysis to support that ITS equipment would continue to adequately meet its designed safety function which was documented in Holtec response to request for technical information (RRTI) #2464-034. However, the licensee discovered that neither change had been fully processed in accordance with SONGS engineering design control process or fully accepted under the Licensee's 10 CFR 50.59/72.48 review process.

These NRC identified issues led to SONGS placing the conditions into their corrective action program (CAP) as action request (AR) 0118-14935. An apparent cause evaluation (ACE) was conducted which reviewed the extent of condition related to vendor changes made to ITS components. The ACE was completed on April 26, 2018. The ACE review documented SONGS's engineering review of 391 Holtec documents, which included 255 construction FCRs, 36 RRTIs, 10 supplier manufacturing deviation reports (SMDRs), and 90 loading FCRs. From that review, the NRC discovered four additional examples where ITS components were modified under Holtec's FCR process without fully following SONGS engineering design change process or SONGS's 10 CFR 50.59/72.48 review process. These items included accept-as-is deviations to one ITS divider shell, two deviations related to the ITS self-hardening subgrade of the ISFSI pad, and one deviation related to the ITS ISFSI top pad surface.

As necessary, the licensee's vendor completed additional calculations for all the components which did not contain rigorous analysis in the original FCR. All the revised calculations and justifications were reviewed by the NRC inspector and were found to contain sufficient engineering analysis to demonstrate the modified ITS components would still be capable of performing their design basis safety functions. Additionally, the design changes were subsequently accepted for implementation by SONGS in accordance with their 10 CFR 50.59/72.48 program.

Section 10 CFR 72.146 (c), "Design Control," states, in part, that the licensee shall subject design changes including field changes, to design control measures commensurate with those applied to the original design.

The licensee's Procedure SO123-XXIV-10.1 titled "Engineering Design Control Process – NECP" Attachment 8, Step 5.5.2, stated, "Design changes to the Dry Cask Storage system are required to be supported by calculations prepared in accordance with this procedure and the 72.48 program."

Contrary to the above, SONGS failed to ensure that design changes or field changes to ITS components were subjected to design control measures commensurate with those applied to the original design. Specifically, a number of field changes to ITS components were not processed in accordance with SONGS engineering design change process with rigorous engineering analysis that demonstrated the changes were acceptable and those changes were not properly accepted for implementation through the Licensee's 10 CFR 50.59/72.48 program.

Consistent with guidance in Section 2.2 of the NRC Enforcement Policy, this violation was dispositioned through the traditional enforcement process. The inspectors used the NRC Enforcement Policy to evaluate the significance of the violation. This violation was determined to have a low safety significance since all the deviations or modification from the original design were found to be acceptable and did not affect the specific components' safety design function or bases. This violation was found to be more than minor since if left uncorrected, it could have the potential to lead to a more significant safety concern. Specifically, failure to adequately control changes and modifications to ITS components could lead to a condition where the appropriate calculation and review was not performed to ensure the component would continue to meet its safety function in accordance with their design basis.

Because the licensee entered the issue into its CAP (AR 0118-14935), the safety significance of the issue was low, the licensee restored compliance, and the issue was not found to be repetitive or willful, this Severity Level IV violation was treated as a Noncited Violation (NCV), consistent with Section 2.3.2.a of the NRC Enforcement Policy (07200041/2017001-001).

2.3 Conclusions

The first loading inspection conducted in January 2018 included 24 hour coverage of the loading operations for the critical tasks associated with the licensee's UMAX loading. Inspectors from NRC Region IV observed operations which included fuel loading, heavy lifts associated with the fuel building crane, welding and nondestructive testing of the canister lid-to-shell weld, hydrostatic pressure testing, FHD drying, helium backfill,

vent/drain port cover welding and nondestructive testing, helium leak testing, radiological surveying, and transport of the loaded transfer cask to the UMAX ISFSI pad.

During the first loading operations, the NRC inspectors identified one violation of 10 CFR 72.146 (c), “Design Control” requirements. The licensee had made modifications to ITS components through the vendor’s (Holtec) corrective action program and did not follow SONGS engineering design change process. The licensee failed to ensure that design changes or field changes to ITS components were subjected to design control measures commensurate with those applied to the original design. The original documentation for the changes was identified to not contain a rigorous engineering analysis that demonstrated the changes were subsequently found to be acceptable and those changes were not properly accepted for implementation through the Licensee’s 10 CFR 50.59/72.48 program. This violation was determined to have a low safety significance since all the deviations or modifications from the original design were found to be acceptable and the changes did not affect the specific components’ safety design function or bases. Because the licensee entered the issue into their corrective action program, the safety significance of the issue was low, the licensee restored compliance, and the issue was not found to be repetitive or willful, this Severity Level IV violation was treated as a NCV, consistent with the NRC Enforcement Policy.

3 Review of 10 CFR 72.212(b) Evaluations (60856)

3.1 Inspection Scope

The programs review inspection conducted on October 9-13, 2017, performed an in depth review of the programs, evaluations, and procedures established to demonstrate that the licensee had met the requirements listed in 10 CFR 72.212 before operation of the UMAX ISFSI.

3.2 Observations and Findings

a. Emergency Planning

The NRC inspectors reviewed the licensee’s Permanently Defueled Emergency Plan (PDEP) to verify and assess the following: (1) the licensee’s emergency action levels (EAL) for accidents that affect the ISFSI; (2) the licensee’s offsite emergency support; and (3) the licensee’s training of employees and conducting periodic drills.

The licensee conducted an evaluation in accordance with 10 CFR 50.54(q) to incorporate the operation of the SONGS UMAX ISFSI into the existing SONGS PDEP. The licensee added definitions and EAL E-HU1.2, “Damage to a loaded canister CONFINEMENT BOUNDARY,” to cover the Holtec spent fuel transport and storage system. The additional EAL threshold for the Holtec system is two times the HI-STORM UMAX technical specifications allowable radiation level on the surface of the VVM or the Holtec transfer cask. The revised PDEP and emergency plan implementing procedures described arrangements with offsite emergency organizations including provisions on how the licensee would conduct periodic drills and training of employees.

b. Fire Protection

The licensee provided an analysis that demonstrated that the site-specific potential for fire and explosions was bounded by the conditions analyzed by the Holtec in accordance with license requirement CoC 72-1040 Appendix B Section 3.4.5. The fire and explosion hazards were analyzed along the haul path and at the UMAX ISFSI in Holtec Report HI-2156567 “Evaluation of Plant Hazards at SONGS,” Revision 2. The explosion hazards analyzed systems and structures which included gasoline tanks, acetylene tanks, lube oil hazards, transformer oil hazards, buildings, and off-site explosions. The assumptions used for the explosion hazards in the report appeared reasonable. No credible explosion hazard was identified at SONGS that exceeded the allowable stress levels identified in the UMAX FSAR which included the overpressure needed to tip over the HI-TRAC VW during transport operations or the structural limits of the closure lids for the UMAX ISFSI. The overpressures for acetylene and gasoline hazards did not exceed the acceptable limits for the UMAX ISFSI or the HI-TRAC VW as long as the specified stand-off distances were met that were incorporated into licensee transportation Procedure HPP-2464-400 “MPC Transfer at SONGS,” Revision 1.

The fire hazards which might affect the cask were identified and reviewed by the licensee. If a fire potential was credible, an evaluation was performed for each postulated hazard to determine if the hazard could exceed the allowable heat input to the cask. Site specific fire hazards included the trailer-mounted fire pump, fixed diesel fire pump, cold and dark standby diesel generator, miscellaneous acetylene tanks, a fuel buggy, and miscellaneous diesel tanks. The assumptions used for the fire hazards in the report appeared reasonable. No credible fire hazard was found to exceed the acceptable heat input to either the HI-TRAC VW or UMAX ISFSI as long as administrative actions included in the licensee Procedure HPP-2464-400 were followed.

During the review of the 10 CFR 72.212 report, the NRC inspectors reviewed the licensee’s analyzed worst case fire during transportation operations to determine whether it was bounded by the analyzed fire in the UMAX FSAR of 50 gallons of diesel fuel from the cask transporter. This evaluation was documented in Holtec report HI-2167264 “Thermal Evaluation of HI-TRAC VW Fire,” Revision 3. The HI-PORT was used to transport the HI-TRAC VW from the fuel handling building to the base area of the UMAX ISFSI. The most limiting scenario was identified to be when the HI-PORT and VCT were next to each other to allow the VCT to engage the HI-TRAC VW to continue transportation to the top of the UMAX ISFSI. Two telescoping man-lifts were also utilized during this transfer event. The combined fire hazard included both fuel tanks of the HI-PORT and VCT, hydraulic fluid from all four pieces of equipment, and the tire rubber associated with the HI-PORT. This fire loading exceeded the 50 gallons of diesel fuel described in the UMAX FSAR. The evaluation determined that the fuel temperature, MPC components, and MPC cavity pressure remained well below the limits established in the UMAX FSAR and the credible fire event did not exceed any FSAR fire accident acceptance criteria. The implementation of this change and associated evaluation was document in a SONGS 10 CFR 72.48 evaluation. Since all the predicted temperatures from the thermal analysis were below the specified temperature limits of short-term events reported in Section 4.5 of the UMAX FSAR, the safety conclusions remained unchanged. The 10 CFR 72.48 evaluation concluded the change did not require NRC approval. The inspectors determined that the 10 CFR 72.48 evaluation was performed adequately.

During the programs review inspection, NRC inspectors reviewed the licensee's Pre-Transport Haul Route Walkdown Checklist (Attachment 8.9) in Procedure HPP-2253-400 to ensure adequate controls were in place to limit combustibles along the haul path and that all fire and explosion hazards had been adequately identified in the reports. No issues were identified by the inspectors relating to the controls implemented to ensure the requirements of the licensee's fire and explosion hazards analyses were met.

c. General License Requirements for 10 CFR 72.212

The SONGS 10 CFR 72.212 Report evaluated the terms, conditions, and specifications in Amendment 2 for the HI-STORM UMAX CoC 72-1040 and documented the conditions as set forth had been met at the SONGS site. Each section of the 10 CFR 72.212 report documented the licensee's compliance with a requirements specified in 10 CFR 72.212(a) through (e). The sections covered topics which included conditions of the license, technical specifications, pad design adequacy, direct radiation, reactor site parameters, written evaluations, physical security, document retention, records, procedures, and program effectiveness.

The NRC inspectors performed a comprehensive review of the Licensee's 10 CFR 72.212 report during the programs review inspection conducted on October 9-13, 2017, and continued the inspection throughout the inspection period with in-office review of the licensee's documentation.

Section 11.0 "Reactor Site Parameters," documented the required written evaluations to verify requirements specified in the Holtec UMAX and FW FSAR and the associated NRC safety evaluation reports were met. The NRC inspectors reviewed these evaluations which related to specific analyses for fires and explosions, tornados, floods, tsunamis and hurricanes, earthquakes, lightning, burial of the ISFSI under debris, environmental temperatures, snow, and collapse of nearby facilities.

The licensee performed a review of the reactor emergency plan, quality assurance program, training program, and radiation protection program and documented the review in Section 15.0, "Program Effectiveness," of the report. Since the TN storage system was already in use, the licensee performed the necessary changes to the programs to incorporate the use of the Holtec UMAX storage system. No issues were identified relating to the NRC's review of the topics discussed above.

d. Quality Assurance

SONGS had a preexisting Generally Licensed 10 CFR Part 72, Subpart G Quality Assurance (QA) program in place for its TN CoC 72-1029 ISFSI. To address transitioning the site from power operations to decommissioning, SONGS developed a decommissioning quality assurance program (DQAP) to support decommissioning activities and to ensure continued oversight of the SONGS ISFSI. The DQAP was SONGS' NRC approved QA program that will be the basis for satisfying the QA requirements of the newly established Holtec HI-STORM UMAX ISFSI and the current TN ISFSI. The NRC inspectors reviewed selected QA activities related to calibrations, receipt inspections, surveillances, and audits.

The Holtec HI-STORM UMAX and HI-STORM FW FSARs identified structures, systems, and components that were ITS and categorized each item into one of three levels (A, B, or C) based on safety significance. The NRC inspectors verified through a review of the SONGS Quality Component List, Rev. 11 that the licensee had incorporated the Holtec HI-STORM UMAX and HI-STORM FW safety designations into their classification scheme along with those of the TN Advanced NUHOMS® System.

The licensee also had a preexisting NRC approved CAP that included the TN Advanced NUHOMS® ISFSI. Holtec, their newest dry fuel storage vendor, also had an NRC-approved CAP. Holtec was handling all fuel loading and radiation protection duties for the pool-to-pad dry fuel storage project for the UMAX ISFSI. After the identification by the NRC of items discussed in Section 2.2.b, Design Control, the licensee made a number of additional changes to ensure that proper evaluation of Holtec condition reports would be performed by SONGS personnel.

e. Radiation Protection

In accordance with 10 CFR 72.104, the licensee provided technical evaluations that demonstrated that the radiation dose from the TN and the UMAX ISFSIs would not exceed 25 mrem per year to the whole body or critical organ or 75 mrem per year to the thyroid of any individual located beyond the owner controlled area. The analyses reviewed by the NRC inspectors also included evaluations that demonstrated no individual would receive a dose greater than the limits specified in 10 CFR 72.106 during any design basis accident at the SONGS site. The UMAX ISFSI was assumed to be fully loaded with fuel characteristics that conservatively exceeded the fuel currently stored in the licensee's spent fuel pools. During loading operations personnel from the SONGS security force established control of public access in areas near the site seawall. The NRC inspectors reviewed site controlled area boundary dose projections in Holtec Report Nos.: HI-2177793, "On-Site and Off-Site Dose Calculations for the SONGS ISFSI," Revision 1, and HI-2156895, "Dose Versus Distance Calculations for the SONGS ISFSI for Compliance with 10 CFR 72," Revision 1. The UMAX accident scenarios were discussed in the Holtec HI-STORM UMAX FSAR.

The UMAX FSAR requires that the radiation protection concept of As Low as Reasonably Achievable (ALARA) be applied to all operations related to dry fuel storage at the SONGS ISFSI. The NRC inspectors verified that SONGS had ALARA policies in place in its radiation protection program through a review of site radiation protection policies and dry fuel loading procedures, including the SONGS Units 2 and 3 Spent Fuel Pool to Pad Project ALARA Plan, Revision 1.

The UMAX FSAR Section 10.3 requires that the shielding effectiveness of the UMAX VVM be assessed after the first MPC canister is placed into the ISFSI. The NRC inspector observed SONGS RP technicians make confirmatory neutron and gamma radiation measurements on the lid of the loaded VVM. The radiation levels present on the VVM lid were consistent with the licensee's site specific Technical Specification 5.3.3 requirements.

The licensee's RP group addressed the external gamma and neutron monitoring of personnel onsite by using electronic dosimeters. The electronic dosimeters used conservative neutron correction factors. This ensured that the real-time monitoring

would provide an over-estimate of actual neutron doses so that these exposures would be managed conservatively. Personnel dose of legal record was measured using thermo-luminescent dosimeters which contained elements sensitive to the presence of neutrons.

The CoC 72-1040 Appendix A Technical Specification 5.3, “Radiation Protection Programs,” included numerous radiation measurement requirements, including the survey locations, and radiation limits. The licensee had incorporated all of the requirements of Section 5.3 in its site procedures and forms. In addition to radiation limits, the technical specification included removable contamination limits on the transfer cask and accessible portions of the MPC. The NRC inspectors verified that SONGS had incorporated those requirements into Procedure HPP-2464-031, “Pool to Pad Certificate of Compliance Radiological Surveys at SONGS,” Revision 0.

f. Records

The inspectors reviewed the licensee procedure SO123-VI-29, “Records Management,” to verify that provisions were in place to maintain records for each cask.

The licensee maintained cask records in accordance with its quality “Procedure SO123-VI-29,” “Records Management,” such that the cask package contained the required information to meet 10 CFR Part 72 requirements for record retention. The inspectors also verified that the licensee incorporated the requirement to register with the NRC no later than 30 days after using the cask to store fuel in Section 7.8.14 of HPP-2464-400, “MPC Transfer.”

3.3 Conclusions

Emergency planning provisions for the UMAX ISFSI had been incorporated into the site’s emergency plan. This included adding a specific EAL for an event damaging loaded UMAX casks.

A fire and explosion hazards analysis had been performed specific to the SONGS UMAX ISFSI. Administrative controls were established to limit the quantity of combustible and flammable liquids around the ISFSI and near the transport path during movement of the canister. The licensee provided calculations demonstrating that the worst case postulated fire event during transportation would not result in a significant increase in the temperature of the spent fuel inside a loaded canister.

The licensee evaluated the bounding environmental conditions specified in the Holtec FSAR and CoC 72-1040 technical specifications against actual site conditions. These included: tornados/high winds, flood, seismic events, tsunamis, hurricanes, lightning, burial of the ISFSI under debris, normal and abnormal temperatures, collapse of nearby facilities, and fires/explosions. The site environmental conditions at SONGS were bounded by the Holtec storage system’s design parameters.

The licensee had implemented their approved reactor facility 10 CFR Part 50 DQAP and CAP for the activities associated with the UMAX ISFSI. Selected QA activities were reviewed related to calibrations, audits, surveillances, and receipt inspections.

The licensee had incorporated keeping radiation exposures ALARA into planning for the cask loading program. Requirements for radiation surveys described in the FSAR and technical specifications had been incorporated into the licensee's procedures for cask loading operations. Projected radiation levels at the ISFSI were calculated for an assumed individual located at the owner controlled area boundary. The analysis demonstrated the dose to this individual would meet the requirements of 10 CFR 72.104.

The licensee was maintaining the 10 CFR Part 72 records in their quality related records system. Records required for retention by 10 CFR 72.174, 10 CFR 72.212, 10 CFR 72.234, and the FSAR had been identified in the licensee's program and were required to be maintained for the life of the ISFSI.

4 Review of 10 CFR 72.48 Evaluations (60857)

4.1 Inspection Scope

The Licensee's 10 CFR 72.48 screenings and evaluations performed to incorporate the use of the UMAX ISFSI were reviewed to determine compliance with regulatory requirements.

4.2 Observations and Findings

a. Safety Evaluations

The licensee had combined the 72.48 screening and evaluation process with the 10 CFR 50.59 process used at the site. Changes to the ISFSI and part 50 facility were processed in accordance with Procedure SO123-XV-4410 "CFR 50.59, 50.82, and 72.48 Program," Revision 21. As part of the programs review inspection, the NRC inspectors reviewed a number of 10 CFR 50.59/72.48 applicability determinations, screens, and one 10 CFR 72.48 evaluation that related to SONGS implementation of the UMAX Storage System.

The licensee completed four larger, nuclear engineering change packages (NECP) to encompass the use of the new UMAX ISFSI. A review was performed by the licensee for each NECP in accordance with 10 CFR 50.59 and 10 CFR 72.48 requirements. Construction of the UMAX ISFSI pad, approach slab, approach ramp, transfer pad, sump area berm, and ISFSI thermal monitoring system was performed under NECP 801372566. The new ISFSI security building was implemented under NECP 801372567 and 801372567. The umbrella NECP that supported implementation of the UMAX system operations for loading spent fuel into a MPC, use of HI-TRAC VW, drying and sealing, transfer of a loaded MPC, and placement at the ISFSI pad was implemented by NECP 801372564. Additionally, the NECP packages were reviewed for potential impacts against the existing TN ISFSI in accordance with 10 CFR 72.48. None of the 10 CFR 50.59/72.48 reviews identified a need for a Part 50 license amendment for the facility.

Section F of the 10 CFR 72.212 report contained a list of changes to the canister storage system licensing basis beyond UMAX FSAR Revision 4. The Holtec engineering change orders (ECO) and SMDRs were identified by the licensee as applicable to the storage system at SONGS. Additional changes to the storage system made by the

vendor would be captured in this list and processed in accordance with SONGS 10 CFR 50.59/72.48 program. Some of these changes were incorporated through the 10 CFR 50.59/72.48 under the previously reviewed NECPs conducted by the licensee. Other changes that occurred after the issuance of the NECPs were accepted by the licensee through standalone or combined screenings with exception of the FCRs previously discussed, for which corrective actions were taken.

The licensee performed one 10 CFR 72.48 evaluation for the implementation of the Licensee's 10 CFR 72.212 report. The 10 CFR 72.48 evaluation identified three areas where implementation of the UMAX storage system at the SONGS site was identified to be different than the descriptions provided in the HI-STORM FW and UMAX FSARs. The three areas related to the combined fire hazard loading (see discussion in Section 3.2.a. of this report), the site's tornado-borne missile differences, and the seismic lateral forces experienced during a DBE when a loaded HI-TRAC VW transfer cask contains a loaded canister in the spent fuel pool.

The SONGS design and licensing basis postulated tornado-borne missiles differed from the missiles addressed in the Holtec FSARs. The licensee's design basis values for rotational wind speed, translational speed, maximum wind speeds, and pressure drop were all less than the values listed in the FSARs. However, the SONGS missiles imparted slightly higher kinetic energy to the various targets for moderate and small missile scope than demonstrated in the FSARs. Since the generic tornado-borne missiles as defined by Holtec do not necessarily bound the site-specific missile parameters for several sites (including SONGS), Holtec prepared a generic report which evaluated the effect of a broader range of postulated site-specific tornado missiles based on the parameters from multiple sites. The generic Holtec Report HI-2135869, "Site-Specific Tornado Missile Analysis for the HI-STORM FW System", Revision 6, re-evaluated the structural impact of the tornado driven missiles on the HI-TRAC and the potential for tip-over and penetration. The applicable tornado-borne missiles evaluated in the generic report bounded all of the SONGS design basis tornado-borne missiles and were summarized in Appendix D of HI-2156567, "Evaluation of Plant Hazards at San Onofre Nuclear Generating Station," Revision 3. The additional evaluations demonstrated that the hypothetical deformations of the UMAX closure lid and impacts to the HI-TRAC VW transfer cask did not compromise the containment boundary of the MPC, locally deform the lid or transfer cask such that the irretrievability of the MPC was threatened, or deform the equipment plastically such that the shielding effectiveness was affected. The evaluation concluded the impacted components had sufficient capacity to withstand the slightly higher loads imparted by the SONGS missiles.

During the site's 10 CFR 72.212 review, the licensee identified that when rigging equipment is being exchanged, for a short period of time, the HI-TRAC VW and loaded MPC is in an unconstrained condition on an intermediate shelf in the spent fuel pool. If a seismic event was to occur during that time frame, the HI-TRAC VW with a loaded MPC could hypothetically fall to the lower level of the spent fuel pool and experience a higher lateral force than previously analyzed by the HI-STORM FW and UMAX FSARs.

The Licensee's 10 CFR Part 50 license and Updated Final Safety Analysis Report had analyzed a potential cask drop from the intermediate shelf to the bottom of the pool as a credible accident. In the past, the licensee had utilized the TN NUHOMS storage system, which contained a lateral side drop evaluation of the TN transfer cask in the TN

FSAR that bounded the site's configuration. The Holtec HI-STORM FW and UMAX FSARs does not contain a side drop analysis for the HI-TRAC VW transfer cask. However, the HI-STORM FW FSAR does contain a tip-over analysis for an MPC inside the HI-STORM overpack storage container.

To evaluate the scenario for this hypothetical accident of the loaded HI-TRAC VW contacting the sides and bottom of the spent fuel pool, the licensee's vendor (Holtec) prepared report HI-2177713 "HI-TRAC Drop in Cask Storage Pool at SONGS", Revision 1. In the report, the licensee demonstrated acceptability of the peak impact deceleration for the HI-TRAC VW scenario at SONGS by comparing those lateral forces to the peak impact deceleration values used to support the 10 CFR Part 71 HI-STAR 190 transport package safety analyses which utilizes the same MPC canister. The licensee's evaluation concluded that the maximum peak lateral deceleration value of the HI-TRAC VW in the pool at SONGS to be 74g's, which was below the HI-STAR 190 side drop evaluation of 85.9g's. Additionally, the MPC and fuel basket evaluated stresses were identified by the licensee to be less than the design basis criteria described in the limiting values from HI-STORM FW FSAR Section 2.2.8. The licensee stated that the same computer software (LS-DYNA) was utilized in all three evaluations (SONGS site specific drop evaluation, HI-STORM FW/UMAX FSAR tip-over evaluation, and HI-STAR FSAR transportation cask drop evaluation).

To utilize this evaluation conducted for the Part 71 HI-STAR 190 transportation license to bound conditions for the storage operations under the 10 CFR Part 72 UMAX license, additional information will need to be submitted by the licensee and evaluated by the NRC to determine if the methodology and implementation of the evaluation through the 10 CFR 72.48 process was appropriate. This item will be tracked as an Unresolved Item (URI) (07200041/2018001-02) until the NRC completes its review of the additional information to determine if the issue of concern potentially constitutes a violation of 10 CFR 72.48 requirements.

4.3 Conclusions

Safety screenings had been performed in accordance with the licensee's procedures and 10 CFR 72.48 requirements. All screenings reviewed were determined to be adequately evaluated. One 10 CFR 72.48 evaluation identified three areas (fire hazards, tornado missiles, and transfer cask drop scenario) where implementation of the UMAX storage system at the SONGS site was identified to be different than the descriptions provided in the HI-STORM FW and UMAX FSARs. All three changes were evaluated by the licensee through the site's 10 CFR 72.48 process to demonstrate the evaluations continued to meet the system's original design basis acceptance criteria listed in the HI-STORM FW and UMAX FSARs. An URI was opened to track the NRC's review of the methodology utilized in the evaluation for a transfer cask drop within the spent fuel pool and determine if the change was acceptable to be performed through the Licensee's 10 CFR 72.48 process.

5 Exit Meeting

The inspectors reviewed the scope and findings of the inspection during a telephonic exit meeting conducted with Mr. Lou Bosch, Plant Manager, and other members of your staff on August 8, 2018.

SUPPLEMENTAL INSPECTION INFORMATION
PARTIAL LIST OF PERSONS CONTACTED

Personnel

A. Bates, Regulatory and Oversight Manager
L. Bosch, Plant Manager
G. Carter, Westinghouse Project Manager
R. Granaas, Reactor Engineering
L. Johnston, Holtec Cask Loading Supervisor
J. Manso, ISFSI Sr. Project Manager
R. McDonald, SCE QC/NDE Oversight
M. Morgan, Regulatory and Oversight
R. Munger, ISFSI Project Manager
J. Smith, Holtec Site Manager
S. Soler, Holtec Site Manager
R. Wagley, Holtec Cask Loading Supervisor

INSPECTION PROCEDURES USED

IP 60854	Preoperational Testing of an ISFSI
IP 60855	Operations of an ISFSI
IP 60856	Review of 10 CFR 72.212(b) Evaluations
IP 60857	Review of 10 CFR 72.48 Evaluations

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

07200041/2017001-01	NCV	Failure to Control Field Design Changes to ITS Components
07200041/2017001-02	URI	10 CFR 72.48 Methodology

Discussed

None

Closed

07200041/2017001-01	NCV	Failure to Control Field Design Changes to ITS Components

LIST OF ACRONYMS

ACE	Apparent Cause Evaluation
ADAMS	Agencywide Documents Access and Management System
AHSM	Advanced Horizontal Storage Module
AISC	American Institute of Steel Construction
ALARA	As Low as Reasonably Achievable
ANSI	American National Standards Institute
AR	Action Request
ASME	American Society of Mechanical Engineers
AWS	American Welding Society
BPVC	Boiler and Pressure Vessel Code
CAP	Corrective Action Program
AR	Action Request
CEC	Cavity Enclosure Container
CFR	Code of Federal Regulations
CMAA	Crane Manufacturers Association of America, Inc.
CoC	Certificate of Compliance
DBE	Design Basis Earthquake
DNMS	Division of Nuclear Material Safety
DSC	Dry Shielded Canister
DSFM	Division of Spent Fuel Management
DQAP	Decommissioning Quality Assurance Program
EAL	Emergency Action Level
EATL	Energy Absorbing Torque Limiter
ECO	Engineering Change Order
FCDB	Fuel Cycle and Decommissioning Branch
FCR	Field Condition Report
FHD	Forced Helium Dehydration
FSAR	Final Safety Analysis Report
FW	Flood and Wind
GTCC	Greater than Class C
HI-PORT	low profile transporter
HI-STORM	Holtec International Storage Module
HI-TRAC VW	transfer cask
IP	Inspection Procedure
ISFSI	Independent Spent Fuel Storage Installation
ITS	Important to Safety
LTS	Leak Test Services
MD	Mating Device
MDA	Mating Device Adapter
MLLW	Mean Lower Low Water
MPC	multi-purpose canister
mrem	milliRoentgen equivalent man
NCV	Noncited Violation
NECP	Nuclear Engineering Change Package
NDE	Nondestructive Examination
NRC	U.S. Nuclear Regulatory Commission
NUHOMS	Nuclear Horizontal Modular Storage
OBE	Operational Basis Earthquake
PCI	PCI Energy Services

PDEP	Permanently Defueled Emergency Plan
QA	Quality Assurance
RP	Radiation Protection
RRTI	Holtec Response to Request for Technical Information
SCE	Southern California Edison
SFP	spent fuel pool
SMDR	Supplier Manufacturing Deviation Report
SONGS	San Onofre Nuclear Generating Station
SSI	Soil Structure Interaction
TN	Transnuclear, Inc.
TS	Technical Specification
UMAX	Underground Maximum Capacity
URI	Unresolved Item
VCT	Vertical Cask Transporter
VVM	Vertical Ventilated Module
X-SAM	Extra Safety and Monitoring

IR 05000206/2017-003, 05000361/2017-003, 05000362/2017-003, AND 07200041/2017-001;
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ADAMS ACCESSION NUMBER: ML18200A400

☒ SUNSI Review By: LEB		ADAMS ☒ Yes ☐ No	☒ Publicly Available ☐ Non-Publicly Available		☒ Non-Sensitive ☐ Sensitive
OFFICE	RIV/DNMS/FCDB	RIV/DNMS/FCDB	NMSS/DFSM/IOB	NMSS/DFSM/IOB	RIV/DNMS/FCDB/BC
NAME	LBrookhart	ESimpson	MDavis	ELove	JKatanic
SIGN	/RA/	/RA/	via email	via email	/RA/
DATE	8/23/18	8/23/18	7/25/18	7/25/18	8/24/18

OFFICIAL RECORD COPY

UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION IV
1600 E. LAMAR BLVD
ARLINGTON TX 76011-4511

October 30, 2015

Mr. Fadi Diya, Senior Vice President
and Chief Nuclear Officer
Union Electric Company
P.O. Box 620
Fulton, MO 65251

SUBJECT: CALLAWAY PLANT - INSPECTION OF THE INDEPENDENT SPENT FUEL STORAGE INSTALLATION (ISFSI) - INSPECTION REPORT 05000483/2015008 AND 07201045/2015001

Dear Mr. Diya:

A team inspection was conducted of your Independent Spent Fuel Storage Installation (ISFSI) between May 19, 2015 and September 1, 2015. The purpose of the inspections were to observe your dry fuel storage preoperational testing activities, to independently assess your readiness to load spent fuel into the ISFSI, and to inspect your initial fuel loading operations. The inspections consisted of six separate inspection trips involving multiple inspectors to observe your dry fuel storage preoperational testing and loading activities. The initial loading of the spent fuel into the first dry fuel storage cask occurred between August 24 - September 1, 2015. The results of the inspections were discussed in an exit with Mr. Mark McLachlan, Senior Director of Engineering and other members of your staff on September 17, 2015.

During the inspections, the NRC staff examined activities conducted under your license as they relate to public health and safety to confirm compliance with the Commission’s rules and regulations, and the conditions of your license. Within these areas, the inspection consisted of selected examination of procedures, representative records, observations of activities, and interviews with personnel. The enclosed report presents the results of these inspections. The inspection determined that you had completed all required activities identified in the Holtec Certificate of Compliance #1040 for use of the Holtec HI-STORM UMAX storage system at your site. No violations of significance were identified and no response to this letter is required.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter, its enclosure, and your response if you choose to provide one, will be made available electronically for public inspection in the NRC Public Document Room or from the NRC's document system (ADAMS), accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html.

To the extent possible, your response should not include any personal, privacy or proprietary information so that it can be made available to the public without redaction.

Should you have any questions concerning this inspection, please contact the undersigned at (817) 200-1191 or Mr. Lee Brookhart at (817) 200-1549.

Sincerely,

/RA/

Ray L. Kellar, P.E., Chief
Repository & Spent Fuel Safety Branch
Division of Nuclear Materials Safety

Dockets: 50-483; 72-104550
License: NPF-30

Enclosure:
Inspection Report 0500483/2015008, 07201045/2015001
w/Attachment: Supplemental Information, Inspector Notes

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/RA/

Ray L. Kellar, P.E., Chief
Repository & Spent Fuel Safety Branch
Division of Nuclear Materials Safety

Dockets: 50-483; 72-1045
License: NPF-30

Enclosure:
Inspection Report 0500483/2015008, 07201045/2015001
w/Attachment: Supplemental Information, Inspector Notes

ADAMS ACCESSION NUMBER: ML15303A348

☒ SUNSI Review By: L.Brookhart		ADAMS ☒ Yes ☐ No		☒ Publicly Available ☐ Non-Publicly Available	☒ Non-Sensitive ☐ Sensitive
OFFICE	RIV/DNMS/RSFS	RIV/DNMS/RSFS	RIV/DNMS/RSFS	NMSS/DSFM/RMB	NMSS/DSFM/IOB
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DATE	10/29/15	10/29/15	10/29/15	10/29/15	10/29/15
OFFICE	NMSS/DSFM/IOB	RIV/DNMS/RSFS			
NAME	JWoodfield	RKellar			
SIGN	***/RA/E-Brookhart***	***/RA/***			
DATE	10/29/15	10/30/15			

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Letter to F. Diya from R. Kellar dated October 30, 2015

SUBJECT: CALLAWAY PLANT - INSPECTION OF THE INDEPENDENT SPENT FUEL STORAGE INSTALLATION (ISFSI) - INSPECTION REPORT 05000483/2015008 AND 07201045/2015001

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U.S. NUCLEAR REGULATORY COMMISSION
REGION IV

Dockets:	50-483 and 72-1045
Licenses:	NPF-30
Report Nos.:	05000483/2015008 and 07201045/2015001
Licensee:	Union Electric Company
Facility:	Callaway Plant and Independent Spent Fuel Storage Installation (ISFSI)
Location:	Junction Highway CC and Highway O, Fulton, Missouri
Dates:	May 19 - 21, 2015, Welding Dry Run June 2 - 4, 2015, MPC Fluid Operations Dry Run June 16 - 18, 2015, MPC Lid to Shell Cutting Dry Run July 13 - 17, 2015, Program Reviews, Fuel Building to ISFSI Dry Run August 3 - 6, 2015, Inside Fuel Building Heavy Loads Dry Run August 24 - September 1, 2015, First Canister Loading Operations
Team Leader:	Lee Brookhart, Senior Inspector, RIV Repository and Spent Fuel Safety Branch
Inspectors:	Eric Simpson, ISFSI Inspector, RIV Gerald Schlapper, Decommissioning Inspector, RIV Clyde Morell, Storage & Transport Safety Inspector, NMSS Jeremy Tapp, Storage & Transport Safety Inspector, NMSS Jon Woodfield, Storage & Transport Safety Inspector, NMSS
Approved By:	Ray L. Kellar, P.E., Chief Repository & Spent Fuel Safety Branch Division of Nuclear Materials Safety

Enclosure

EXECUTIVE SUMMARY

Callaway Plant and Independent Spent Fuel Storage Installation
NRC Inspection Report 50-483/2015-08 and 72-1045/2015-01

The NRC team conducted an extensive evaluation of Callaway's program for the safe handling and storage of spent fuel at their UMAX ISFSI, observed the pre-operational training demonstrations, and observed the loading of the first spent fuel cask system. The Callaway Plant had selected the Holtec Certificate of Compliance #1040, HI-STORM UMAX cask storage system for use at their site's Independent Spent Fuel Storage Installation (ISFSI). This system consisted of the Multi-Purpose Canisters (MPC-37) that store 37 pressurized water reactor fuel assemblies in a below grade Vertical Ventilated Module (VVM). Callaway had constructed the UMAX ISFSI to hold 48 MPC-37s within the VVMs at the site. The ISFSI location includes space to eventually expand the UMAX to accommodate an additional 96 VVMs, if the site feels that it will be required in the future. The ISFSI was licensed by the NRC under the general license provisions of 10 CFR Part 72, Subpart K. The licensee planned to load six canisters for placement within the UMAX ISFSI during fall of 2015, of which the first canister loading was observed by the NRC.

This inspection report covers six separate inspections conducted between May 19 and September 1, 2015. During the dry run demonstrations and loading activities the inspectors verified compliance with licensing documents: Holtec Certificate of Compliance No. 72-1040 and Technical Specifications, Amendment 0; the UMAX Final Safety Analysis Report (FSAR), Revision 2; the FW FSAR, Revision 3; the NRC's Safety Evaluation Report for 72-1040, Amendment 0. Callaway developed a pre-operational test plan which consisted of five dry run demonstrations encompassing the pre-operational testing and training exercises required by License Condition 8 of the Holtec Certificate of Compliance. The demonstrations were conducted under the observation of the NRC. Twenty- three technical areas were reviewed during the inspections including such topical areas as the overhead crane requirements, loading operations, fuel verification, radiological programs, quality assurance, heavy loads, training, welding, fire protection and others. Subsequent to the site visits, an extensive in-office review was performed of documents provided by the Callaway staff. This effort involved the review of licensee reports, procedures, calculations, training documents, test results, personnel qualification records, safety evaluations, and condition reports to support the conclusion that the licensee had developed and implemented a comprehensive program to support ISFSI activities.

During the inspections, the licensee completed the demonstrations related to the operations of equipment and the implementation of procedures to verify that all operations required by the technical specifications could be performed safely. The programs review conducted by a NRC team of six inspectors, concluded that the licensing requirements related to dry cask storage had been adequately incorporated into the site's programs and procedures. During the various pre-operational demonstrations and first loading, the Callaway workers demonstrated a comprehensive understanding of the technical requirements related to the loading and operations of an ISFSI. Callaway's first cask was placed within the site's UMAX ISFSI on September 1, 2015.

Details related to the technical areas reviewed during this inspection are provided as Attachment 2 "Callaway Inspector Notes" to this inspection report. The following provides a summary of the observations of this inspection.

Canister Drying/Inerting

- Forced helium dehydration dryness limits established in Technical Specification A.3.1.1.1 and Table 3-1 had been incorporated into the licensee's procedures. The licensee planned to use the forced helium dehydration system for drying all canisters loaded at the site. Operation of the forced helium dehydration system was demonstrated during the pre-operational dry run exercises.

- Helium backfill pressure requirements established in Technical Specification A.3.1.1.2 and Table 3-2 had been incorporated into the licensee's procedures.

Crane Design

- The licensee had evaluated their fuel building 125-ton crane against the criteria in NUREG 0554, ASME NOG-1, and CMAA Spec #70-2010 and found the crane to meet the criteria for a single failure proof crane.

- Specific aspects of the crane which included: the bridge and trolley brakes, main hoist safety devices, emergency stop features, crane two-block protection, and dual rope reeving system met the requirements of NUREG 0554 and NUREG 0612.

Crane Inspection

- The 125-ton fuel building crane was subjected to a daily inspection, performed prior to use, that satisfied the requirements of ASME B30.2, Section 2-2.1.2 "Frequent Inspection." On an annual basis the crane was subjected to a more rigorous inspection that met the requirements of ASME B30.2, Section 2-2.1.3 "Periodic Inspection."

- A performance test was completed at Callaway after the new 125-ton trolley and hoist were installed. The site test included hoist raising/lowering at all speeds, trolley travel in both directions at all speeds, bridge travel in both directions at all speeds, and testing of all safety devices.

- The crane's hook was inspected annually as required by ASME B30.10, Sections 10-1.4.2 through 10-1.4.6.

Crane Load Testing

- Callaway's new 125-ton crane's trolley and hoist were dynamically load tested to 100% of the rated load and statically loaded to 125% after installation within the fuel building in June 2014.

- The fuel building 125-ton crane utilized a 150-ton hook which was subjected to a 200% hook load test of 300 tons in February 2014.

Crane Operation

- The maximum weight the 125-ton crane would lift during the cask loading campaign was 123.5 tons when lifting the HI-TRAC VW transfer cask containing the MPC-37 canister loaded with spent fuel out of the spent fuel pool.

- Callaway's qualification requirements for crane operators were consistent with the requirements listed in ASME B30.2.

- The licensee had the ability to manually lower the load and manually move the bridge and trolley if an emergency occurred. These provisions had been incorporated into a licensee procedure.

- The licensee's procedures required brake checks and specified a minimum travel height when lifting the HI-TRAC VW.

Dry Run Demonstration

- The licensee successfully completed all the required pre-operational tests specified by License Condition #8 of the Certificate of Compliance. This included welding, drying, and backfilling of a mock-up canister and the simulated unloading of a sealed canister. A weighted canister was used to demonstrate heavy load activities inside the fuel building, transport between the fuel building and the UMAX ISFSI, and movement back into the fuel handling building for unloading purposes. Inside fuel building dry runs included placement of an empty MPC and HI-TRAC VW into the spent fuel pool and movement of a dummy fuel assembly into the MPC. Additionally, Holtec performed a dry run to demonstrate the removal of the canister lid welds, for unloading purposes, at Holtec Manufacturing Division (HMD).

Emergency Planning

- Emergency planning provisions for the ISFSI had been incorporated into the site-wide emergency plan. This included adding a specific emergency action level for an event damaging a loaded cask. Part 50 emergency action levels applicable to the ISFSI included fires, security threats, and events involving a radiological release from a canister.

Fire Protection

- A Fire Hazards Analysis had been performed specific to the Callaway UMAX ISFSI. Administrative controls were established to limit the quantity of combustible and flammable liquids around the ISFSI and near the transport path during movement of the canister.

- Site specific fire and explosion hazards had been evaluated to determine the effect on the UMAX ISFSI and to confirm that the location of the ISFSI, location of the transport route, and the design of the transportation equipment was adequate. Several nearby facilities were evaluated that included diesel tanks and pipes, lube oil tanks, gasoline tanks, hydrogen tanks, and delivery trucks of flammable liquids.

- Additional fire analysis was required for the use of the tracked vertical cask transporter (VCT), the low profile transporter (HI-PORT), and ancillary equipment to account for the fire loading due to all vehicles' fuel contents and their hydraulic fluids. Holtec provided calculations that showed the postulated fire involving all the equipment would not result in a significant increase in the temperature of the spent fuel inside a loaded canister when being transported.

Fuel Selection/Verification

- For the initial loading campaign, the licensee planned to load only intact fuel assemblies that met the requirements of Technical Specification Appendix B, Section 2.1, Section 2.3, and the associated tables. The fuel assemblies selected for the first canister met the limits for length, width, weight, irradiation cooling time, average burn-up, cladding, decay heat, and fuel enrichment.

- The licensee planned to load fuel in the canisters using the regionalized fuel loading concept allowed in Technical Specification Appendix B, Section 2.3 and Figure 2.3-1. For the initial loading campaign, the licensee selected the option to load cooler spent fuel into the outer canister locations to provide shielding to the hotter fuel assemblies that were placed in the inner locations of the canister.

- The licensee had established provisions for independent verification of the correct loading of spent fuel assemblies into the canister. This included use of an underwater camera to view the fuel assemblies' serial numbers.

General License Requirements

- The licensee evaluated the bounding environmental conditions specified in the Holtec FSAR and Certificate of Compliance No. 1040 Technical Specifications against the conditions at the site. This included: tornados, flood, seismic events, hurricanes/high winds, lightning, burial of the ISFSI under debris, snow/ice, normal and abnormal temperatures, and fires/explosions. The site environmental conditions at Callaway were bounded by the Holtec cask design parameters except for fire, explosions, and tornado driven missiles. Separate analyses showed that the site's ISFSI and dry cask storage transportation operations could withstand Callaway's site specific tornado driven missile, worst postulated fire event, and pressures due to an explosion.

- Projected radiation levels at the ISFSI were calculated for an assumed individual located at the owner controlled area boundary to determine the dose to this individual. The analysis assumed that the ISFSI was fully loaded with all 48 canisters in the UMAX ISFSI with fuel characteristics that bounded the UMAX design basis. The calculation concluded that the dose to any individual would be a small fraction of the regulatory limits. The calculated doses were well below the 10 CFR 72.104 limit of 25 mrem/year.

- The licensee performed an evaluation of the Part 50 reactor programs that could be impacted by the addition of an ISFSI. The evaluation included a review of the radiation protection program, emergency planning program, quality assurance program, training program, reactor technical specifications, and the Part 50 license. Revisions to the programs to incorporate the ISFSI were identified and implemented. None of the

changes required an amendment to the plant's Part 50 operating license or technical specifications.

- The Holtec Certificate of Compliance and FSAR had been reviewed by the licensee to verify that the design basis for the Holtec system and the conditions and requirements in the Certificate of Compliance and FSAR were met.

- Callaway had developed procedures for controlling all work associated with cask handling, loading, movement, surveillance, maintenance, and testing. Procedures had been developed specific to the ISFSI activities. Numerous other procedures developed for the Part 50 reactor programs were being adequately applied to the ISFSI program.

Heavy Loads

- The licensee had incorporated the special requirements related to the ISFSI project into the site heavy loads programs and procedures. Crane operators interviewed were knowledgeable of the special handling requirements related to the spent fuel casks.

- Special lifting device height limits and temperature restrictions during movement of the canisters had been incorporated into the licensee's procedures consistent with the requirements in the Certificate of Compliance and FSAR.

- A safe loads path had been identified and analyzed for moving the spent fuel from the spent fuel pool. Provisions were established in procedures to prevent the crane operators from moving the loaded canister outside the boundaries of the safe load path while in the fuel building.

- The adequacy of the vertical cask transporter (VCT) for the expected weight of a loaded transfer cask and the ability of the transporter to safely perform downloading operations at the UMAX ISFSI was verified by NRC inspectors. The VCT was static load tested to 125% of its rated capacity and was given a 100% performance load test prior to fuel loading operations.

Loading Operations

- Requirements in the FSAR related to pre-operational inspections and annual maintenance of equipment were being implemented through the licensee's procedures.

- Technical specifications and FSAR requirements related to spent fuel boron concentration, fuel cladding not being exposed to air, handling of damaged fuel containers, and time-to-boil limits were implemented in the licensee's procedures.

- During the loading of the first canister beginning August 24, 2015, the NRC provided 24-hour coverage of the loading operations for all the critical tasks. This included fuel movement, heavy lifts of the loaded canister, radiation surveys of the loaded transfer cask and storage cask, welding of the lid and port cover plates, Forced Helium Dehydration (FHD) drying, helium backfill of the canister, and transportation of the canister into the UMAX ISFSI. The first canister was placed into the UMAX ISFSI on September 1, 2015.

Non-Destructive Examination

- The requirements to perform helium leak testing of each canister was incorporated into the licensee's procedures. The helium leak testing equipment used during the initial loading operations was verified to meet the minimum sensitivity level specified in ANSI N14.5.

- A review of the visual and liquid penetrant examination specialists' qualifications identified that they were properly qualified as a Level II examiners.

- The welding contractor, utilized by Callaway for dry cask storage operations, implemented visual and liquid penetrant examination procedures that met all the applicable requirements from ASME Section III, Section IV, and the FSAR in regards to non-destructive examination of welds.

Pressure Testing

- The requirements for canister hydrostatic testing had been incorporated into the licensee's procedure and were consistent with the requirements of ASME Section III Subsection NB, Article NB-6000.

- The hydrostatic testing sequence and criteria described in the FSAR had been incorporated into the licensee's procedures.

Quality Assurance

- The licensee had implemented their approved reactor facility Part 50 quality assurance program for the activities associated with the ISFSI. Selected Quality Assurance (QA) activities were reviewed related to calibrations, operating status, receipt inspections, QA surveillances, and QA audits.

- The FSAR identified structures, systems, and components that were important to safety and categorized each item into one of three levels (A, B, or C) based on safety significance. The licensee incorporated Holtec's safety designations into their classification procedure used to determine the level of quality control to place on the items.

- A corrective action program that documented issues and classified problems according to their impact on quality and safety was being effectively used by the licensee. Selected condition reports were reviewed to verify adequate resolution of the issues.

Radiation Protection

- The licensee had incorporated As Low As Reasonably Achievable (ALARA) planning into the dry cask loading program. This included developing reasonable dose goals, utilizing lessons learned from other sites, and conducting radiation pre-job briefings that identified expected radiological conditions for the different work evolutions.

- Requirements for radiological and contamination surveys described in the FSAR and technical specifications had been incorporated into the licensee's health physics

program for the loading of the canisters. This included decontamination of the transfer cask and canister lid, and performing required surveys of the transfer cask and loaded VVM.

- The licensee incorporated proper neutron dose consideration into the health physics monitoring program for neutrons that would be present around the canister when empty of water. This consideration included the use of appropriate personnel dosimetry that could measure neutron doses and applying a correction factor based on survey readings obtained during the loading campaign.

Records

- The licensee was maintaining the ISFSI records in their quality related records system. Records required for retention by 10 CFR 72.174, 10 CFR 72.212, 10 CFR 72.234, and the FSAR had been identified in the licensee's program as required records for retention. The records were required to be maintained for five years after the transfer of the fuel from the ISFSI.

Safety Reviews

- Changes to the site related to the construction and operation of the ISFSI were being evaluated in accordance with 10 CFR 72.48 and 10 CFR 50.59 requirements. No issues were identified during the review of selected safety screenings and full safety evaluations.

Slings

- The slings used for downloading the MPC and other slings utilized throughout the campaign met the requirements of NUREG 0612. Operations required dual/redundant slings that had a rated capacity of twice the sum of the static and dynamic loads. All slings' proof loading tests were found to meet the requirements of ASME B30.9.

- The sling inspection program complied with ASME B30.9 in regards to daily sling inspections, annual sling inspections, and proof loading.

Special Lifting Devices

- The licensee's special lifting device program complied with ANSI N14.6 in regards to stress design, prior to use inspections, and 300% proof loadings for the lift yoke, lift yoke extensions, HI-TRAC VW Lift Lugs, MPC lift cleats, and the VCT lift links.

Storage Operations

- The inspection of the VVM outlet and inlet air ducts to be free from blockage was placed in the licensee's procedures to be performed daily as required by Technical Specification A.3.1.2.

Unloading

- The licensee procured the equipment and developed procedures to perform gas sampling if a canister was required to be unloaded. The licensee demonstrated the gas sampling process to the NRC during the dry run demonstrations.

- Canister re-flooding for unloading was demonstrated to the NRC during the dry run demonstrations. The procedure controlling canister re-flooding contained all of the applicable requirements from the FSAR and the technical specifications.

Welding

- Requirements for hydrogen monitoring during welding of the canister lid had been incorporated into the licensee's procedures.

- All welding procedures contained the required essential, non-essential, and supplemental variables specified in ASME Section IX for gas tungsten arc welding.

- The welding procedure's qualification test coupons all satisfactorily passed the required bend and tension tests to qualify the welding procedures and thus the lid to shell weld.

- The welder's performance qualification test records were reviewed and documented that the welders had met the qualification testing requirements for manual and machine welding of the canister lid. The testing requirements complied with the requirements of ASME Section IX.

SUPPLEMENTAL INSPECTION INFORMATION

PARTIAL LIST OF PERSONS CONTACTED

Licensee Personnel

R. Brumnet, Consultant, QC Inspector
C. Emerson, Consultant, Radiation Protection
D. Daugherty, QC Inspector
S. Ewens, ISFSI Project Manager
R. Lutz, Lead Project Engineer
M. McLachlan, Senior Director Engineering
J. Mcinvale, Consultant, Project Engineer
T. Pettus, Project Engineer
L. Ptasznik, Regulatory Affairs
G. Roesner, Consultant, Project Engineer
D. Shafer, Consultant, Regulatory Affairs
E. Stewart, Senior Reactor Operator

Holtec International

B. Bates, Field Supervisor
A. Brown, Project Engineer
D. Burns, Welder/Machine Operator
T. Card, Project Manager
T. Ciesielski, Welder/Machine Operator
N. DeDominicis, HMD Manager of Projects
L. Johnson, Field Supervisor
L. Kinney, Site Manager
S. Roland, Operations Manager
J. Sloane, Field Supervisor
S. Soler, Director Site Services Manager
R. Tindal, Project Manager
T. Witt, Regulatory Affairs

PCI Group

R. Campbell, Welder
D. O'Conner, NDE Inspector
L. Hobson, Welder
J. Meyers, Welding Supervisor
L. Vice, NDE Inspector
J. York, Welder

INSPECTION PROCEDURES USED

IP 60854.1	Preoperational Testing of ISFSIs at Operating Plants
IP 60855.1	Operations of an ISFSI at Operating Plants
IP 60856	Review of 10 CFR 72.212(b) Evaluations
IP 60857	Review of 10 CFR 72.48 Evaluations

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

<u>Opened</u>

None

<u>Discussed</u>

None

<u>Closed</u>

None

LIST OF ACRONYMS

abs	absolute
ADAMS	Agencywide Documents Access and Management System
ALARA	As Low As Reasonably Achievable
ANSI	American National Standards Institute
ASHRAE	American Society of Heating Refrigerating and Air-Conditioning Engineer
ASME	American Society of Mechanical Engineers
ASTM	American Society for Testing and Materials
atm	atmosphere
AWS	automatic welding system
BP	Burnable Poison Rod Assembly
CAL	Calculation
CAR	Corrective Action Request
CARS	Corrective Action Request System
CEC	Cavity Enclosure Container
cc/sec	cubic centimeters per sec
CFR	Code of Federal Regulations
CLP	Cask Loading Pit
cm	centimeter
CMAA	Crane Manufacturers' Association of America
CMTR	certified materials test report
CoC	Certificate of Compliance
CWP	Cask Washing Pit
DBE	Design Basis Earthquake
DFC	damaged fuel container
DNMS	Division of Nuclear Material Safety
dpm	disintegrations per minute
EAL	emergency action level
EPD	electronic pocket dosimeter

EPP	emergency plan procedure
EPRI	Electrical Power Research Institute
F	Fahrenheit
FHD	forced helium dehydration
fps	feet per second
FSAR	Final Safety Analysis Report
g	acceleration due to gravity
gpm	gallons per minute
GQP	General Quality Procedure
GTAW	gas tungsten arc welding
GWD/MTU	Giga Watt Day per Metric Ton Uranium
GWS	General Welding Standard
HI	Holtec International
HI-PORT	Low Profile Transporter
HI-TRAC VW	Holtec Transfer Cask
HMD	Holtec Manufacturing Division
HMSLD	helium mass spectrometer leak detector
HSP	Holtec Standard Procedure
HPP	Holtec Project Procedure
HVAC	heating, ventilating, and air-conditioning
ICRP	International Commission Radiation Protection
ICA	item control area
IP	Inspection Procedure
IPTE	Infrequently Performed Test Evolution
ISFSI	Independent Spent Fuel Storage Installation
ISG	Interim Staff Guidance
ITS	important to safety
IWRC	independent wire rope core
kW	killo-watt
lbs	pounds
LCO	limiting condition for operation
m/sec	meters per second
MPC	multi-purpose canister
mrem	MilliRoentgen Equivalent Man
MSLD	mass spectrometer leak detector
MWD/MTU	Mega Watt Day per Metric Ton Uranium
NDE	non-destructive examination
NOG	Nuclear Overhead and Gantry (Cranes)
NEI	Nuclear Energy Institute
NIST	National Institute of Science and Technology
NITS	not important to safety
NRC	Nuclear Regulatory Commission
NSA	Neutron Source Assembly
NUREG	US Nuclear Regulatory Commission Regulation
OCA	owner controlled area
OSL	optically stimulated luminescence
OQAM	Operating Quality Assurance Manual
PCI	Westinghouse Electric Company Welding Service
PII	personally identifiable information
PM	preventative maintenance
PQR	procedure qualification record

PS	Purchase Specification
psig	pounds per square inch gauge
PT	liquid penetrant exam
PWR	pressurized water reactor
QA	quality assurance
QC	quality control
RCCA	Rod Cluster Control Assemblies
RERP	Radiological Emergency Response Plan
RIV	Region 4
RVOA	removable valve operator assembly
RWP	radiation work permit
S/N	serial number
SER	Safety Evaluation Report
SFP	spent fuel pool
SNM	special nuclear material
SNT-TC	Society for Non-Destructive Testing-Technical Committee
Spec	Specification
SSC	structures, systems, and components
SSE	safe shutdown earthquake
TAL	threaded anchor location
TEPC	Tissue Equivalent Proportional Chamber
TLD	thermo-luminescent dosimetry
TP	Thimble Plug
TS	technical specification
U-235	Uranium 235
UFSAR	Updated Final Safety Analysis Report
UFSAR-SA	Updated Final Safety Analysis Report Site Addendum
UFSAR-SP	Updated Final Safety Analysis Report Standard Plant
VCT	vertical cask transporter
VVM	Vertical Ventilated Module
VT	Visual Testing
WOPQ	welder operator performance qualification
WPQ	welder performance qualification
WPS	welding procedure specification
ZPA	Zero Period Accelerations
Zr	zirconium based fuel cladding

ATTACHMENT 2

CALLAWAY INSPECTOR NOTES

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CALLAWAY INSPECTOR NOTES

Category: Canister Drying/Inerting **Topic:** Dryness Levels

Reference: CoC 1040, Tech Spec A.3.1.1.1 and Table 3-1 Amendment 0

Requirement When using the forced helium dehydration (FHD) system for moisture removal, the gas temperature exiting the demoisturizer shall be 21 degrees F or less, for 30 minutes or more, or the gas dew point exiting the canister shall be 22.9 degrees F or less, for 30 minutes or more.

Observation: The procedures used at Callaway for canister sealing exceeded the dryness requirements established by the Certificate of Compliance (CoC). The spent fuel canisters used at Callaway used the forced helium dehydration (FHD) system to ensure that fuel dryness levels were adequately obtained. Certificate of Compliance (CoC) 1040 Technical Specification (TS) surveillance requirement 3.1.1.1 and Table 3-1 required that the gas temperature exiting the demoisturizer be less than or equal to 21 degrees F for 30 minutes or more, or that the gas dew point exiting the canister be less than or equal to 22.9 degrees F for 30 minutes or more. These TS requirements were placed in the Holtec procedure HPP-2253-300 for canister sealing. However, the procedures used at Callaway conservatively exceeded the dryness levels required by the Holtec CoC TSs. The procedure used at Callaway required the temperature leaving the demoisturizer to be no more than 16 degrees F for 30 minutes or more. The gas dew point sensor was utilized by the system earlier in the FHD process at Callaway, but the licensee used the demoisturizer exiting temperature for the TS surveillance to ensure the system was dry.

Documents Reviewed: a) Holtec Procedure HPP-2253-300, "MPC Sealing at Callaway," Revision 7

Category: Canister Drying/Inerting **Topic:** Forced Helium Dehydration System (FHD)

Reference: CoC 1040, Appendix A, Section 3.6.1, Table 3-1 Amendment 0

Requirement For canisters containing one or more fuel assemblies with burnup values greater than 45 GWD/MTU, forced helium dehydration must be used for canister drying. For all other canisters, either forced helium dehydration or vacuum drying may be used for canister drying, unless MPC-37 is loaded using Figure 2.3-12.

Observation: Use of the forced helium dehydration system was planned for all canisters loaded at Callaway. The first canister contained assemblies with a burnup of greater than 45 GWD/MTU. The licensee demonstrated the use of the forced helium dehydration system during the dry run conducted the week of June 2-4, 2015. Procedure HPP-2253-300 provided instructions for use of the system. Use of the forced helium dehydration system was always required in Procedure HPP-2253-300, and as such, the criteria in Technical Specification A.3.6.1 was not specified in the procedure.

Documents Reviewed: a) Holtec Procedure HPP-2253-300, "MPC Sealing at Callaway," Revision 7

Category: Canister Drying/Inerting **Topic:** Helium Backfill Pressure

Reference: CoC 1040, Tech Spec A.3.1.1.2, Table 3-2, et al. Amendment 0

Requirement For the MPC-37 canister with Standard Fuel, backfill pressure shall be as follows: for

cask heat loads less than or equal to 33.88 kW, using canister loading option depicted in Figure 2.3-1 helium backfill shall be equal to or greater than 41.0 psig up to 44.2 psig. The pressure range is at a reference temperature of 70 degree F.

Observation: All canisters loaded at Callaway were restricted to the loading option depicted in CoC Appendix B, Figure 2.3-1 "UMAX MPC-37 Permissible Heat Load Chart 1 for Long-term Storage for Short and Standard Fuel" and a helium backfill pressure range of equal to or greater than 41.0 psig up to 44.2 psig (at 70 degree F) per procedure ETP-ZZ-04020.

Helium backfill pressure requirements were incorporated into Procedure HPP-2253-300 consistent with the requirements in TS A.3.1.1.2 and Table 3-2. Procedure HPP-2253-300, Section 7.10, "FHD Helium Backfill Operation," provided instructions for the helium backfill operations. Step 7.10.7 of Procedure HPP-2253-300 used Attachment 8.6 "FHD Helium Backfill Pressure Chart" to determine the acceptable pressure range at different temperatures based on a 70 degree F reference temperature.

Documents Reviewed: a) Certificate of Compliance No. 1040, "For the HI-STORM UMAX Cask Storage System," Amendment 0; b) Callaway Procedure ETP-ZZ-04020, "Fuel Selection and Cask Loading for Dry Cask Storage," Revision 0; c) Holtec Procedure HPP-2253-300, "MPC Sealing at Callaway," Revision 7

Category: Crane Design **Topic:** Bridge and Trolley Brakes

Reference: NUREG 0554, Section 5.1 Published May 1979

Requirement Bridge and trolley control and holding brakes should be: a) rated at 100% of maximum drive torque that can be developed at the point of application; b) adjusted with one brake in each system leading the other; and c) automatically actuate on interruption of power and overspeed. The holding brakes should be designed so that they cannot be used as foot-operated slowdown brakes. Drag brakes should not be used.

Observation: The bridge and trolley brakes on the single-failure proof crane met the NUREG requirements. The bridge/trolley control and the holding brakes were capable of applying a counter torque that would be 100% of maximum drive torque that could be developed at the point of application. The trolley and bridge brakes were provided in arrangements in accordance with the Crane Manufacturers Association of America (CMAA) Spec #70-2010 with one brake system leading the other. The bridge and trolley motors were provided with spring set, electronically released holding brakes that would automatically apply when power became interrupted. The design of the bridge and trolley holding brakes were such that they cannot be used as a foot-operated slowdown brake and drag brakes were not used for the bridge or trolley drives.

Documents Reviewed: a) American Crane and Equipment Corporation Document REP-2150-003, "NUREG 0554/0612 Compliance/Safety Analysis Report for the Cask Handling Crane Upgrade Holtec International/Ameren's Callaway Energy Center," Revision 0

Category: Crane Design **Topic:** Drum Safety Devices

Reference: NUREG 0554, Section 4.2 Published May 1979

Requirement The hoist drum should be provided with structural and mechanical safety devices to limit

its drop during a shaft or bearing failure. The devices should prevent disengaging from the holding brake.

Observation: The single-failure proof crane at Callaway met the design requirement. The main hoist drum retaining devices were steel structures, which ensured that a shaft or bearing failure would not allow the main hoist drums to disengage from the brakes.

Documents Reviewed: a) American Crane and Equipment Corporation Document REP-2150-003, "NUREG 0554/0612 Compliance/Safety Analysis Report for the Cask Handling Crane Upgrade Holtec International/Ameren's Callaway Energy Center," Revision 0

Category: Crane Design **Topic:** Emergency Stop Feature

Reference: NUREG 0554, Sections 3.3, 6.1, and 6.6 Published May 1979

Requirement An emergency stop feature should be installed at the control station. For cranes remotely operated using radio control stations, a second emergency stop feature should be provided at ground level to remove power from the crane, independent of the controller. Cranes that use more than one control station should be provided with electrical interlocks that permit only one control station to be operated at a time.

Observation: The requirement for an emergency stop button to exist on both the remotely operated overhead crane belly box (both the pendant and radio remote controls) and in another redundant location accessible from the ground floor was visually verified by NRC inspectors during preoperational dry-runs at the Callaway Nuclear Station. The presence and locations of the emergency stop buttons was also detailed in the American Crane NUREG 0554 Compliance Report.

Documents Reviewed: a) American Crane and Equipment Corporation Document REP-2150-003, "NUREG 0554/0612 Compliance/Safety Analysis Report for the Cask Handling Crane Upgrade Holtec International/Ameren's Callaway Energy Center," Revision 0

Category: Crane Design **Topic:** Seismic Events During Cask Movement

Reference: NUREG 0554, Section 2.5 Published May 1979

Requirement The crane should be designed to retain control of and hold the load, and the bridge and trolley should be designed to remain in place on their respective runways with their wheels prevented from leaving the tracks during a seismic event.

Observation: The crane was designed to retain control of and hold the load during a seismic event. The trolley was designed to remain in place on their respective runways with their wheels prevented from leaving the tracks during a seismic event. The new trolley, including main hoist, was designed and fabricated to meet CMAA Specification #70-2010, ASME NOG-1-2004, and NUREG 0554 requirements. The main-hoist was single-failure-proof and designed to maintain control of the maximum rated load during the seismic spectra specifications from the plant's Part 50 Updated Final Safety Analysis Report (UFSAR). This analysis was documented in Calculation CAL-21250-SE-001 "Crane Structural Analysis & Trolley Evaluation" and in the licensee's 50.59 screen MP 14-0014 "Dry Fuel Storage Licensing and Operations Documentation (Heavy Loads Review)."

Documents Reviewed: a) Calculation CAL-21250-SE-001, "Crane Structural Analysis & Trolley Evaluation," Revision 1; b) 50.59 Screen MP 14-0014, "Dry Fuel Storage Licensing and Operations Documentation (Heavy Loads Review)," Revision 0

Category: Crane Design **Topic:** Seismically Induced Load Swing

Reference: NUREG 0554, Section 2.5; Reg Guide 1.29 Published May 1979

Requirement The maximum critical load plus operational and seismically induced pendulum and swing load effects on the crane should be considered in the design of the trolley and should be added to the trolley weight for the design of the bridge.

Observation: The maximum critical load plus the swing load effects due to a seismically induced pendulum was included in the design calculations associated with Callaway's 125-ton fuel building crane. Licensee Calculation CAL-21250-SE-001 "Crane Structural Analysis & Trolley Evaluation" included the crane with the maximum crane load suspended from the trolley at several locations along the bridge and at several heights within the fuel building. A linearly elastic, static finite element analysis was used to obtain operational load analyses and environmental seismic member forces. The computer program used for this calculation was the SAP2000 Version 15.1.0, which was permitted by the Part 50 program's UFSAR. The analyses determined that the trolley members and connections satisfied all allowable stress criteria set forth in CMAA #70-2010 and ASME NOG-1.

Documents Reviewed: a) Calculation CAL-21250-SE-001, "Crane Structural Analysis & Trolley Evaluation," Revision 1; b) 50.59 Screen MP 14-0014, "Dry Fuel Storage Licensing and Operations Documentation (Heavy Loads Review)," Revision 0

Category: Crane Design **Topic:** Single Failure Proof Crane

Reference: NUREG 0554, Section 1.0 Published May 1979

Requirement When reliance for the safe handling of critical loads is placed on the crane system itself, the system should be designed so that a single failure will not result in the loss of the capability of the system to safely retain the load.

Observation: The original cask handling crane in Callaway's fuel handling building was a non-single-proof 150-ton crane. In early 2014, Callaway replaced the 150-ton non-single failure proof crane/trolley with a new 125-ton single-failure proof crane/trolley from American Crane and Equipment Corporation. This new 125-ton single-failure proof crane was designed to meet the requirements of NUREG-0554, NUREG 0612, CMAA #70-2010, ASME B30.2, and ASME NOG-01.

Documents Reviewed: a) American Crane and Equipment Corporation Document REP-2150-003, "NUREG 0554/0612 Compliance/Safety Analysis Report for the Cask Handling Crane Upgrade Holtec International/Ameren's Callaway Energy Center," Revision 0; b) 50.59 Screen MP 14-0014, "Dry Fuel Storage Licensing and Operations Documentation (Heavy Loads Review)," Revision 0

Category: Crane Design **Topic:** Two-Block Protection

Reference: NUREG 0554, Section 4.5 Published May 1979

Requirement The complete hoisting system should have the required strength to resist failure during two-blocking. As an alternative, a system of upper travel limit switches may be used to prevent two-blocking. The system should include two independent travel limit devices of different designs and activated by separate mechanical means. These devices should de-energize the hoist drive motor and the main power supply. The auxiliary hoist, if used for critical lifts, should also be equipped with two independent travel limit switches to prevent two-blocking.

Observation: The requirement for two-block protection was met at Callaway through the design of the main hoist that allowed the crane to two-block without cutting or crushing the wire ropes or causing permanent deformation or damage to the crane. In addition, redundant and independent upper travel limit switches to prevent two-blocking from ever occurring were also incorporated into the design. As such, the Cask Handling Crane used at Callaway had multiple paths of two-block protection.

Documents Reviewed: a) American Crane and Equipment Corporation Document REP-2150-003, "NUREG 0554/0612 Compliance/Safety Analysis Report for the Cask Handling Crane Upgrade Holtec International/Ameren's Callaway Energy Center," Revision 0

Category: Crane Design **Topic:** Wire Rope Breaking Strength

Reference: NUREG 0554, Section 4.1 Published May 1979

Requirement The maximum load (including static and inertia forces) on each individual wire rope in the dual reeving system with the maximum critical load attached should not exceed 10% of the manufacturer's published breaking strength.

Observation: The 125 ton single failure proof crane's wire rope met the NUREG 0554 requirement. The maximum load rating for the main hoist was 125-tons or 250,000 lbs. The wire ropes used on the main hoist were rated for 195,800 lbs. The main hoist utilized two Python Power 9V EEIPS wire ropes that were each 1-1/8" nominal diameter and length of 775 ft through an 8-part reeving system. Therefore the stress in each individual wire rope from the maximum load would be 15,625 pounds (250,000/16). Ten percent of the manufacturer's published breaking stress for the wire rope was 19,580 lbs. Therefore the maximum load on the wire rope at maximum rated load was 15,625 lbs, which is less than 10 percent of the published breaking strength of 19,580 lbs, meeting the NUREG requirement.

Documents Reviewed: a) UNIROPE Test Certificate No 191303 Item #5 Rel #2, Dated 12/12/2013; b) UNIROPE Test Certificate No 191303 Item #2 Rel #2, Dated 12/10/2013

Category: Crane Design **Topic:** Wire Rope Configuration

Reference: NUREG 0554, Section 4.1 Published May 1979

Requirement A dual rope reeving system with individual attaching points and a load balancing system will permit either rope system to hold and transfer the critical load without excessive shock in case of failure of the other rope system. The dual reeving system may be a

single rope from each end of the drum terminating at one of the blocks or equalizer with provisions for equalizing beam-type load and rope stretch, with each rope designated for the total load. Alternatively, a 2-rope system may be used from each drum or separate drums using a sheave equalizer or beam equalizer or other combination that provides two separate and complete reeving systems.

Observation: The single-failure proof crane at Callaway met the NUREG requirement. The 125-ton crane at Callaway used two drums with two ropes, with a balanced dual reeving system with each rope terminating on the drum it originated on.

Documents Reviewed: a) American Crane and Equipment Corporation Document REP-2150-003, "NUREG 0554/0612 Compliance/Safety Analysis Report for the Cask Handling Crane Upgrade Holtec International/Ameren's Callaway Energy Center," Revision 0

Category: Crane Inspection **Topic:** Crane Inspection - Frequent

Reference: ASME B30.2; Section 2-2.1.2 Published 1976

Requirement Cranes in regular use shall be subjected to a frequent crane inspection monthly during normal service, weekly to monthly during heavy service, and daily to weekly during severe service. The frequent inspection points shall include: a) operating mechanisms for proper operation daily; b) all limit switches should be checked at the beginning of each work shift by inching, or running at slow speeds, each motion into its limit switch; c) leakage in lines, tanks, valves, pumps, and other parts of the air or hydraulic systems; d) hooks for cracks, more than 15% of normal throat opening, or more than 10 degrees of twist; e) hook latches for proper operation; f) hoist ropes including end clamps; and g) the rope reeving system.

Observation: NRC inspectors verified via procedure review that the frequent cask handling crane inspections performed at Callaway would look for the following items: proper operation of operating mechanisms, limit switch functionality, fluid leaks, hook and latch wear and operation, wire rope end connections, and proper spooling on the drum. The 125-ton crane was daily inspected in accordance with ASME B30.2, Section 2-1.1.2.

Documents Reviewed: a) Callaway Procedure OTS-KE-00016, "Operation of the Cask Handling Crane," Revision 22; b) Job Order #15500182.500, "Semi-Annual Crane Inspection HKE14 (Mechanical)," Revision 3

Category: Crane Inspection **Topic:** Crane Inspection - Periodic

Reference: ASME B30.2; Section 2-2.1.3 Published 1976

Requirement Cranes in regular use shall be subjected to a periodic crane inspection annually during normal and heavy service, and quarterly during severe service. The periodic inspection includes checking for: a) deformed, cracked or corroded members; b) loose bolts or rivets; c) cracked or worn sheaves and drums; d) worn, cracked or distorted pins, bearings, shafts, gears, and rollers; e) excessive brake system wear; f) load, wind, and other indicators over their full range for any significant inaccuracies; g) gasoline, diesel, electric, or other power plants for improper performance; h) excessive drive chain sprocket wear and chain stretch; and i) deterioration of controllers, master switches, contacts, limit switches and pushbutton stations.

Observation: The periodic crane inspection was performed as required. Callaway had implemented three different job orders to inspected the 125-ton fuel building crane. Job Order #14502408.500 "Annual Crane Maintenance and Inspection (HKE14)," contained annual mechanical inspection requirements and was completed on June 20, 2015. Job Order #14502408.321 "Annual Crane Inspection HKE14 (Electrical)," contained electrical inspection requirements and was completed June 25, 2015. Job Order #15500182.500 "Semi-Annual Crane Inspection HKE14 (Mechanical)," contained additional mechanical inspection requirements and was completed June 20, 2015. All ASME B30.2 inspection requirements were captured between the three inspection procedures and were being performed on an annual or semi-annual basis.

Documents Reviewed: a) Job Order #14502408.500, "Annual Crane Maintenance and Inspection (HKE14)," Revision 2; b) Job Order #14502408.321, "Annual Crane Inspection HKE14 (Electrical)," Revision 1; c) Job Order #15500182.500, "Semi-Annual Crane Inspection HKE14 (Mechanical)," Revision 3

Category: Crane Inspection **Topic:** Crane Operational Testing

Reference: ASME B30.2; Sect 2-2.2.1 Published 1976

Requirement Prior to initial use, all new, reinstalled, extensively repaired, or modified cranes shall have the following functions tested: (a) lifting and lowering, (b) trolley travel, (c) bridge travel, (d) limit switches, and (e) locking and safety devices. The trip setting of the hoist limit devices shall be determined by tests with an empty hook traveling in increasing speeds up to the maximum speed. The actuating mechanism of the limit device shall be located so that it will trip the device under all conditions in sufficient time to prevent contact of the hook or load block with any part of the trolley or crane.

Observation: All ASME B30.2 operational testing requirements were tested during the site acceptance test performed in June of 2014 per Procedure REP-21250-007.

Documents Reviewed: a) Procedure REP-21250-007, "American Crane & Equipment Corporation Site Load Test," Revision 1

Category: Crane Inspection **Topic:** Hoist Overload Testing

Reference: NUREG 0554, Section 8.3; NUREG 0612, C-4, (9) Published 1979/1980

Requirement If the hoisting system is designed with adequate strength to resist failure during load hang-up, the hoisting system should be tested by securing the load-block-attaching points to a fixed anchor and applying the maximum critical load. Alternately, if a load cell system, a motor current-sensing device, or a mechanical load-limiting device is provided to prevent load hang-up, the device(s) should be tested to verify operability.

Observation: Testing was performed that met the NUREG requirement. A two-block test of the main hoist was performed during factory functional testing at American Crane & Equipment Corporation's factory on March 27, 2014 per REP-21250-004 and REP-21250-005. The overweight limits on the main hoist were calibrated and tested during the factory load testing. The proper operation of the mechanical slip clutch, which was present to mitigate the effects of two-blocking or load hang-up event, was verified for the 125-ton main hoist.

Documents Reviewed: a) Procedure REP-21250-004, "American Crane & Equipment Corporation Factory Functional Test Procedure," Revision 1; b) Procedure REP-21250-005, "American Crane & Equipment Corporation Factory Load Test Procedure," Revision 1

Category: Crane Inspection **Topic:** Hook Inspections - Frequent/Periodic

Reference: ASME B30.10, Sections 10-1.4.2 and 10-1.4.6 Published 1975

Requirement Hooks shall be inspected monthly during normal service, weekly to monthly during heavy service and daily to weekly during severe service. Hooks should be inspected for: a) distortion such as bending, twisting or increased throat opening; b) wear; c) cracks, severe nicks, or gouges; d) latch engagement, damaged or malfunctioning latch (if provided); and e) hook attachment and securing means. Hooks having any of the following deficiencies shall be removed from service unless a qualified person approves their continued use and initiates corrective action: a) cracks; b) wear exceeding 10% of the original sectional dimension; c) bend or twist exceeding 10 degrees from the plane of an unbent hook; and d) an increase in throat opening of 15% (for hooks without latches).

Observation: NRC inspectors verified via procedure review that Callaway met the hook inspection requirements of ASME 30.10, "Hooks," in its frequent (daily) or periodic (semi-annual) crane hook inspections. This criteria included inspections of the hook attachment points, for distortion, cracks, and wear, and for crane hook latch engagement problems. These requirements were met for both the 125-ton main hoist as well as the 5-ton auxiliary hoist.

Documents Reviewed: a) Callaway Procedure OTS-KE-00016, "Operation of the Cask Handling Crane," Revision 22; b) Job Order #15500182.500, "Semi-Annual Crane Inspection HKE14 (Mechanical)," Revision 3

Category: Crane Inspection **Topic:** Wire Rope Inspection - Frequent

Reference: ASME B30.2, Section 2-2.4.1 (a) Published 1976

Requirement All ropes shall be visually inspected once each working day.

Observation: NRC inspectors verified via procedure review that Callaway met the requirements of ASME B30.10, Section 2-2.4, "Rope Inspection, Replacement, and Maintenance." According to the procedure used at Callaway, the wire rope is to be inspected prior to each shift as part of an operational check. The procedure required that the wire rope be inspected for broken/damaged wire, excessive wear, kinks, distortion, corrosion, and problems with the end clips/rope attachment points.

Documents Reviewed: a) Callaway Procedure OTS-KE-00016, "Operation of the Cask Handling Crane," Revision 22

Category: Crane Inspection **Topic:** Wire Rope Replacement Criteria

Reference: ASME B30.2, Section 2-2.4.2 Published 1976

Requirement Conditions such as the following should be sufficient reason for questioning continued use of the rope, or increasing the frequency of inspection: a) twelve randomly distributed broken wires in one lay; b) wear of one-third of the original diameter of outside

individual wires; c) kinking, crushing, bird caging or any other damage resulting in distortion of the rope structure; d) evidence of heat damage; and e) reduction in diameter in excess of nominal.

Observation: The wire rope on the 125-ton crane was inspected to the ASME B30.2, Section 2-2.4.2 criteria semi-annually per Job Order #15500182.500, "Semi-Annual Crane Inspection HKE14 (Mechanical)." The last semi-annual inspection was conducted on June 20, 2015. Steps 5.1.9 from the Job Order contained the ASME B30.2 Section 2-2.4.2 wire rope inspection requirements.

Documents Reviewed: a) Job Order #15500182.500, "Semi-Annual Crane Inspection HKE14 (Mechanical)," Revision 3

Category: Crane Load Testing **Topic:** Cold Proof Testing

Reference: NUREG 0554, Section 2.4; NUREG 0612, C-2 (8) Published 1979/1980

Requirement Minimum operating temperatures for the crane should be specified to reduce the possibility of brittle fracture of the ferritic load-carrying members of the crane. The minimum temperature can be determined by: 1) a drop weight test per ASTM E-208, 2) a Charpy test per ASTM A-370, or 3) a 125% cold proof test. If the crane is made of low alloy steel such as ASTM A514, cold proof testing should be done. If cold proof testing is omitted, the default minimum crane operating temperature is 70 degrees F. For crane operation at temperatures below 70 degrees F, cold proof testing must be performed and the ambient temperature at which the testing is conducted becomes the minimum crane operating temperature.

Observation: The 125-ton crane's operating temperature was set per procedure requirements to exceed 70 degrees F. The MPC loading procedure in step 5.11 required operators to ensure cask handling crane temperature to be greater than 70 degrees F.

Documents Reviewed: a) Callaway Procedure APA ZZ-00365-Addendum L, "Callaway Lifting Operations," Attachment 10, Table 1, "Fuel Handling Crane Data," Revision 18; b) Holtec Procedure HPP-2253-200, "MPC Loading at Callaway," Revision 9

Category: Crane Load Testing **Topic:** Dynamic Load Testing (100%)

Reference: NUREG 0554, Section 8.2 Published May 1979

Requirement After the 125% static load test, the crane should be given a full performance test with 100% of the maximum critical load attached, for all speeds and motions for which the system is designed. This should include verifying all limiting and safety control devices.

Observation: A full performance test with 100% of the maximum critical load attached was performed for all speeds and motions for which the system was designed. The 100% site load test was completed on June 4, 2014. Procedure REP-21250-007 page 12 of 41, documented that the licensee satisfactorily performed the test with a load of 253,100 lbs. All speeds and motions were tested as well as the limiting and safety control devices.

Documents Reviewed: a) Procedure REP-21250-007, "American Crane & Equipment Corporation Site Load Test," Revision 1

Category: Crane Load Testing **Topic:** Hook Load Testing

Reference: NUREG 0554, Sect 4.3; ASME B30.10, Sect 10-1.1.2 Published 1979/1975

Requirement A 200% static load test should be performed for each load-attaching hook. For a duplex (sister) hook, the proof load shall be shared by the two sisters unless the hook is designed for unbalanced loading. Measurements of the geometric configuration of the hooks should be made before and after the test and the acceptance criteria is no permanent increase in throat opening in excess of 0.5% or 0.010 inches (0.25 mm). The load testing should be followed by a nondestructive examination that should consist of volumetric and surface examinations to verify the soundness of fabrication and ensure integrity of the hooks.

Observation: A static hook load test of 200% was performed on the 125-ton main hoist of the fuel building crane. A 150-ton hook was purchased for the new fuel building crane. McKissick Certificate of Conformance No 840554 documented that the 150-ton hook was proof load tested to 600,000 lbs on February 19, 2014.

Documents Reviewed: a) McKissick Certificate of Conformance Test Certificate No 840554, "Certificate of Test and Examinations of Chains, Rings, Hooks, Shackles, Swivels, and Pulley Blocks," dated 03/12/14

Category: Crane Load Testing **Topic:** NDE Exams Following Cold-Proof Testing

Reference: NUREG 0554, Section 2.4 and 2.6 Published May 1979

Requirement Following the 125% cold-proof testing, a nondestructive examination of the welds whose failure could result in the drop of a critical load should be performed. If any of these weld joint geometries would be susceptible to lamellar tearing, the base metal at the joints should be nondestructively examined. Nondestructive examination of critical areas should be repeated at 4-year intervals or less.

Observation: Cold proof testing was performed at Callaway following the 125-ton crane's static load test at 125% of the rated capacity. Non-destructive examination was performed on the critical welds identified by the licensee. The welds examined were documented in Attachment A-1 "Table of Required Inspections, Test, and Documentation" and in Attachment A-2 "Identification of Critical Welds & Brittle Fracture Concerns for Existing Bridge Structure for the Cask Handling Crane Upgrade" of American Crane and Equipment Corporation Document REP-2150-003.

Documents Reviewed: a) Callaway Procedure APA ZZ-00365-Addendum L, "Callaway Lifting Operations," Attachment 10, Table 1, "Fuel Handling Crane Data," Revision 18; b) Holtec Procedure HPP-2253-200, "MPC Loading at Callaway," Revision 9; c) American Crane and Equipment Corporation Document REP-2150-003, "NUREG 0554/0612 Compliance/Safety Analysis Report for the Cask Handling Crane Upgrade Holtec International/Ameren's Callaway Energy Center;" Revision 0

Category: Crane Load Testing **Topic:** Rated Load Marking

Reference: NUREG 0554, Section 8.5; ASME B30.2, Sect 2- Published 1976

Requirement The rated load shall be marked on each side of the crane and, if the crane has more than one hoisting unit, each hoist shall have its rated load marked on it or on its load block.

This marking shall be legible from the ground or floor.

Observation: The rated load of the fuel building overhead crane at Callaway was marked on each side of the crane and on each of the hoists. The fuel building crane was rated as 125 tons. One other hoist was located on the crane as well, a 5-ton hoist.

Documents Reviewed: a) NUREG 0554, "Single Failure Proof Cranes for Nuclear Power Plants," published May 1979

Category: Crane Load Testing **Topic:** Static Load Testing (125%)

Reference: NUREG 0554, Section 8.2 Published May 1979

Requirement The crane should be static load tested at 125 percent of the maximum critical load. The test should be conducted at all positions generating maximum strain in the bridge and trolley structures and other positions as recommended by the designer or manufacturer.

Observation: A static load test at 125% of the maximum critical load attached was performed at Callaway on June 4, 2014. Procedure REP-21250-007, page 23, documented that the licensee satisfactorily performed the test with a load of 318,200 lbs.

Documents Reviewed: a) Procedure REP-21250-007, "American Crane & Equipment Corporation Site Load Test," Revision 1

Category: Crane Operation **Topic:** Brake Test Prior to Lift

Reference: ASME B30.2, Section 2-3.2.3 (g) Published 1976

Requirement The operator shall check the hoist brakes at least once each shift if a load approaching the rated load is to be handled. This shall be done by lifting the load a short distance and applying the brakes.

Observation: This requirement had been adequately incorporated into the lifting procedures at Callaway. Callaway's lifting operations procedure included step 4.2.1.t that described the "lift and hold test" for all lifts performed with the main hoist during fuel building operations. In this step, all loads were lifted to clear the ground or securing device for underwater lifts and the load was held long enough to listen for unusual sounds, check the load on the load indicator, review and inspect the rigging, and to ensure that the load is stable. A similar step was included in the lifting and rigging program procedure in step 4.2.3. The crane operating procedures used at Callaway were compliant with the brake testing criterion of ASME B30.2.

Documents Reviewed: a) Callaway Procedure APA ZZ-00365, Addendum L, "Callaway Lifting Operations," Revision 18; b) Callaway Procedure APA-ZZ-00365, "Callaway Lifting and Rigging Program," Revision 26

Category: Crane Operation **Topic:** Height Limit During Cask Movement

Reference: No Reference Provided

Requirement For single failure proof cranes, the cask height during movement should be sufficiently high to allow for engaging of the brakes during an uncontrolled descent before the load would impact the floor.

Observation: Callaway had placed this requirement into their loading procedures. Procedure HPP-2253-200 contained a Note right above Step 7.7.40 that stated, "The HI-TRAC VW must be maintained greater than 9" above the 2047'-6" deck while not centered over the Cask Loading Pit or Cask Wash Down Pit." The nine inch lift height was conservatively above the manufacturer's recommendations of sufficient height to ensure the main hoist brakes would engage during an uncontrolled lowering event.

Documents Reviewed: a) Holtec Procedure HPP-2253-200 "MPC Loading at Callaway," Revision 9

Category: Crane Operation **Topic:** Hoist Limit Switch Tested Each Shift

Reference: ASME B30.2, Section 2-3.2.4 (a) Published 1976

Requirement At the beginning of each shift, the operator shall try out the upper limit device of each hoist under no-load. Care shall be exercised. The block shall be inched into the limit or run in at a slow speed.

Observation: The Callaway procedure for the cask handling crane listed all of the inspection areas and steps to be performed for daily preoperational testing. Step 5.6.4 (e) of the crane operation procedure tests the upper limit cut-off of the main hoist. This demonstrated that Callaway met the requirements of ASME B30.2, section 2-3.2.4, which required that at the beginning of each shift, the operator shall try out the upper limit device of the hoist under no load.

Documents Reviewed: a) Callaway Procedure OTS-KE-00016, "Operation of the Cask Handling Crane," Revision 21

Category: Crane Operation **Topic:** Qualification For Crane Operator

Reference: ASME B30.2, Sections 2-3.1.2 and 2-3.1.6 Published 1976

Requirement Qualification to operate a cab operated or remote operated crane, requires the operator to pass a written or oral examination and a practical operating examination specific to the type of crane to be operated unless able to furnish evidence of previous qualification and experience. In addition, the operator shall: a) have vision of at least 20/30 Snellon in one eye and 20/50 in the other with or without corrective lenses; b) be able to distinguish colors regardless of their position; c) have sufficient hearing capability for the specific operation with or without hearing aids; d) have sufficient strength, endurance, agility, coordination and reaction speed for the specific operation; e) have evidence of not having physical defects or emotional instability that would interfere with crane operation; and f) not be subject to seizures, loss of control, or dizziness.

Observation: NRC performed an onsite inspection of crane operator qualifications at the Callaway site on August 6, 2015. NRC inspectors reviewed the qualification documents while onsite and returned them to Callaway so that there were no problems with handling personally identifiable information (PII). NRC confirmed the qualifications of the crane operators at Callaway through its review of vertical cask transporter (VCT), HI-PORT, Overhead Crane, Cask Handling Crane Upgrade, and National Commission for the Certification of Crane Operator (NCCO) training course records, among others. All of the crane operators were verified as meeting the requirements listed in ASME B30.2, sections 2-3.1.2 and 2-3.1.6.

Documents Reviewed: a) Various personnel training records for crane operators at the Callaway site (PII)

Category: Crane Operations **Topic:** Maximum Weight of Canister

Reference: FSAR 1040, Section 9.0, Table 3.2.1 Revision 2

Requirement The maximum weight of the transfer cask containing the canister filled with water and fuel (including dynamic loads) that will be lifted by the crane is to be verified to be within the crane's rated capacity.

The handling weights for the Holtec storage system components are provided in Tables 3.2.1 of the FSAR. The user shall implement controls to ensure all critical set points (e.g., lift weights) do not exceed design limits of the specific equipment.

Observation: The licensee had verified that the weight of the Holtec storage system components would not exceed the rated load of the Callaway fuel building crane. Holtec Report No. HI-2146011, Table 7.2 documented the maximum lifted weight during the performance of a HI-STORM UMAX System, loaded at Callaway, would be during the removal of the HI-TRAC VW transfer cask from the spent fuel pool with the neutron water jacket full of water (Case 2 in Table 7.2). Case 2 of Table 7.2 showed a total lift weight of 247,088 pounds (123.5 tons), which was less than the 125-ton rating of the fuel building crane. The individual component weights shown in Table 7.2 were consistent with those in Holtec FSAR.

Documents Reviewed: a) Holtec Report Number HI-2146011, "Cask Handling Weights at Callaway," Revision 1; b) Holtec Report HI-2115090, "Final Safety Analysis Report (FSAR) on the HI-STORM UMAX Canister Storage System," Revision 2

Category: Crane Operations **Topic:** Provisions For Manual Operation

Reference: NUREG 0554, Sections 3.4; 4.9 Published May 1979

Requirement A crane that has been immobilized because of failure of controls or components while holding a critical load should be able to hold the load or set the load down while repairs or adjustments are made. This can be accomplished by inclusion of features that will permit manual operation of the hoisting system and the bridge and trolley transfer mechanisms by means of appropriate emergency devices.

Observation: The American Crane & Equipment Corporation supplied 125-ton single-failure proof crane for Callaway's fuel building included features that permitted manual operation of the hoisting system and the bridge and trolley transfer mechanisms. These features were tested on June 12, 2014 during the Callaway 100% site acceptance test of the crane per Procedure REP-21250-007. Callaway's Engineering group also developed a site procedure that could be utilized to manually lower a suspended load in the event of a crane malfunction. Callaway Procedure OTS-KE-00016, Attachments 4-6 contained steps to allow manual operation to lower a load, manually operate the bridge, and manually operate the trolley.

Documents Reviewed: a) Procedure REP-21250-007, "American Crane & Equipment Corporation Site Load Test," Revision 1; b) Callaway Procedure OTS-KE-00016, "Operation of the Cask Handling Crane," Revision 22

Category: Dry Run Demonstration **Topic:** Fuel Loading and Verification Demonstration

Reference: CoC 1040, Condition 8.c, d Amendment 0

Requirement The dry run shall include selection and verification of specific fuel assemblies to ensure type conformance and the loading of specific assemblies into the canister (using a dummy fuel assembly), including appropriate independent verification.

Observation: NRC inspectors observed fuel movement operations in the spent fuel pool (SFP) and assembly placement into multiple MPC slots during Callaway ISFSI dry-run #4 demonstration, August 4-6, 2015. Callaway fuel movers demonstrated the ability to access difficult to reach locations of the SFP and most restrictive areas of the MPC, as situated in the cask loading pit. A walk-through of how the independent verification process would be controlled was also demonstrated during the dry run. Fuel handlers at Callaway fully satisfied the CoC 1040 condition during this preoperational demonstration.

Documents Reviewed: a) Callaway Procedure OTS-KE-00012, "Spent Fuel Pool Bridge Crane," Revision 33

Category: Dry Run Demonstration **Topic:** MPC Pressure Test, Drying, and Helium Backfill

Reference: CoC 1040, Condition 8.f Amendment 0

Requirement The dry run shall include pressure testing, draining, moisture removal (by vacuum drying or forced helium dehydration, as applicable), and helium backfilling. A mockup may be used for this dry-run exercise.

Observation: NRC inspectors observed and documented MPC draining, pressure testing, force helium dehydration, and helium backfill activities during Holtec's second dry run demonstration at Callaway on June 2-4, 2015. A part-size mockup was used for these demonstrations, so the time that it took for operations was greatly abbreviated. Holtec successfully demonstrated pressure testing requirements of the MPC and met all of the required dryness and helium backfill levels just as would be required for a full-sized MPC with actual fuel loaded inside.

Documents Reviewed: a) Holtec Procedure HPP-2253-300, "MPC Sealing at Callaway," Revision 7

Category: Dry Run Demonstration **Topic:** MPC Removal from Spent Fuel Pool

Reference: CoC 1040, Condition 8.e Amendment 0

Requirement The dry run shall include remote installation of the canister lid and removal of the canister and transfer cask from the spent fuel pool or cask loading pool.

Observation: NRC inspectors observed the remote installation of the MPC lid and removal of the MPC and HI-TRAC transfer cask from the cask loading pit (inside the spent fuel pool) to the cask wash-down area of the fuel building. This activity was demonstrated during dry-run #4, August 4-6, 2015. The licensee utilized Procedure HPP-2253-200 to perform the dry run. Callaway fully satisfied this preoperational criteria in the Holtec UMAX Certificate of Compliance 1040.

Documents Reviewed: a) Holtec Procedure HPP-2253-200, "MPC Loading at Callaway," Revision 9

Category: Dry Run Demonstration **Topic:** MPC Transfer to UMAX

Reference: CoC 1040, Condition 8.g Amendment 0

Requirement The dry run shall include transfer cask of the MPC from the transfer cask to the UMAX VVM.

Observation: Callaway demonstrated MPC transfer operations to the UMAX VVM during dry run #3 on July 14-16, 2015. NRC inspectors verified successful transfer of a dummy MPC-37 canister to the UMAX VVM at Callaway. Holtec followed procedure HPP-2253-400.

Documents Reviewed: a) Holtec Procedure HPP-2253-400, "MPC Transfer at Callaway," Revision 6

Category: Dry Run Demonstration **Topic:** MPC Welding and NDE

Reference: CoC 1040, Condition 8.f Amendment 0

Requirement The dry run shall include canister welding and non-destructive examination (NDE) of the canister lid.

Observation: NRC inspectors observed MPC closure welding and the non-destructive testing operations (visual, liquid dye penetrant, and helium leak check) of those welds during the Holtec UMAX dry-run #1 on May 19-21, 2015. Holtec and its welding contractor PCI successfully demonstrated welding of the MPC closure lid to shell weld, port cap covers, and closure ring during its first dry run demonstration for NRC. In addition, NDE of those welds and helium leak testing operations were observed during this dry run. Callaway met this CoC 1040 requirement.

Documents Reviewed: a) PCI Project Instruction, PI-CNSTR-OP-CAL-H-01, "Closure Welding of Holtec Multi-Purpose Canisters - UMAX," Revision 0; b) PCI General Welding Standard - 1 (GWS-1), Revision 0; c) PCI General Quality Procedure, GQP-9.2, "High Temperature Liquid Penetrant Examination and Acceptance Standards for Welds, Base Materials, and Cladding," Revision 8; d) PCI General Quality Procedure, GQP-9.7, "Liquid Penetrant Examination and Acceptance Standards for Welds, Base Materials, and Cladding," Revision 11; e) PCI Procedure GQP-9.6, "Visual Examination of Welds," Revision 14

Category: Dry Run Demonstration **Topic:** Placement of MPC in Spent Fuel Pool

Reference: CoC 1040, Condition 8.a, b Amendment 0

Requirement The dry run shall include moving the MPC and the transfer cask into the spent fuel pool or cask loading pool and preparing the HI-STORM UMAX cask system for fuel loading.

Observation: NRC inspectors observed movement of the MPC and HI-TRAC transfer cask into the spent fuel pool cask loading pit during Callaway's ISFSI dry-run #4 demonstration, August 4-6, 2015. The licensee utilized Procedure HPP-2253-200 to perform the dry run. Callaway fully satisfied the Holtec UMAX CoC 1040 condition during this preoperational demonstration.

Documents Reviewed: a) Holtec Procedure HPP-2253-200, "MPC Loading at Callaway," Revision 9

Category: Dry Run Demonstration **Topic:** Unloading a Canister

Reference: CoC 1040, Condition 8.h Amendment 0

Requirement The dry run shall include HI-STORM UMAX system unloading, including flooding the canister cavity and removing canister lid welds. A mockup may be used for this dry-run exercise.

Observation: NRC observed the demonstration of reflooding of a simulated previously loaded MPC during the second dry-run demonstration at Callaway on June 2-4, 2015. Reflooding of the canister was demonstrated after the drying and backfill dry run was completed. Placement of the MPC and HI-TRAC transfer cask back into the spent fuel pool for unloading was demonstrated during the final dry-run on August 4-6, 2015. These operations were successfully demonstrated as part of Callaway's dry run activities leading up to the first loading during the week of August 24, 2015.

Callaway's contractor, Holtec, completed the MPC cutting dry run on July 16-18, 2015. Holtec performed the MPC lid to shell weld cutting dry run at Holtec Manufacturing Division (HMD) located in Turtle Creek, PA. Callaway personnel were in attendance for this dry run and performed oversight activities throughout the operations. NRC inspectors from headquarters also observed the cutting dry run. Holtec utilized Procedure HPP-2253-500 to perform the cutting demonstration. The cutting activities included boring through the cover plate and the MPC vent/drain port covers. Utilization of a cutting machine to remove the lid to shell weld, while purging the area under the lid with argon, monitoring for hydrogen during the duration of the cutting demonstration, and removal of the MPC lid after the cutting was successful. Ultimately, the ability to core bore through the closure ring and vent/drain port cover plates and cut the MPC lid to shell weld was demonstrated. As a result of the demonstration, a number of procedure enhancements to the implementing procedure HPP-2253-500 were documented on Callaway CAR 201501626, Action 7. In addition, a Holtec International Bulletin (HIB)/Lessons Learned addressing shim installation during lid placement prior to lid to shell welding was to be issued and was also documented on the same CAR. All procedure enhancements were successfully completed prior to Callaway's first loading campaign.

Documents Reviewed: a) Holtec Procedure HPP-2253-500, "MPC Unloading at Callaway," Revision 0 and 7

Category: Emergency Planning **Topic:** Emergency Drills

Reference: 10 CFR Part 50, App E, Section F.1 Published 2015

Requirement The emergency program shall provide for the training of employees and exercising, by periodic drills, of radiation emergency plans to ensure that employees are familiar with their specific emergency response duties.

Observation: No emergency drills had been conducted at the site, specific to the ISFSI. Training had been provided to site personnel on the new emergency action level scheme in April 2015, which included the emergency action levels for the ISFSI. Site personnel conducted drills annually related to plant operations that included all aspects of an emergency response that would be applicable to the ISFSI including fire, medical response, and

radiological drills.

Documents Reviewed: a) Callaway Plant Radiological Emergency Response Plan (RERP), Revision 46

Category: Emergency Planning **Topic:** ISFSI Emergency Plan

Reference: 10 CFR 72.32(c) Published 2015

Requirement For an ISFSI that is located on the site of a nuclear power plant licensed for operation, the emergency plan required by 10 CFR 50.47 shall be deemed to satisfy the requirements of this section.

Observation: The licensee was using their Part 50 emergency plan for the ISFSI. Radiological Emergency Response Plan, Table 4-1, "Emergency Action Level (EAL) Classification Matrix," had incorporated the ISFSI and identified an unusual event as the classification for a problem at the ISFSI. Procedure EIP-ZZ-00101, Addendum 2 had incorporated the ISFSI into the emergency action level scheme and identified damage to a loaded cask confinement boundary as the initiating event. This damage could come from different issues which included damage due to a dropped or tipped over cask, explosion at the ISFSI, projectile damage, fire damage, or natural phenomena affecting a cask would classify as an unusual event. Several other EALs could be related to the ISFSI. Due to the type of welded canisters in use, there were no specific EALs related to the ISFSI for radiological releases. However, the site area emergency classification could be reached if radiation levels offsite were measured in excess of 100 mrem. If the levels exceeded 1,000 mrem, a general emergency would be declared. These readings would be based on surveys taken in the field. For a security event, a hostile action in the owner controlled area was an alert. A hostile action within the protected area, including the ISFSI protected area, would be a site area emergency.

Documents Reviewed: a) Callaway Plant Radiological Emergency Response Plan (RERP), Revision 46; b) Callaway Procedure EIP-ZZ-00101 Addendum 2, "Emergency Action Level Technical Bases Document, " Revision 9

Category: Fire Protection **Topic:** Fire Accident Response

Reference: FSAR 1040, Section 12.2.1.3 Revision 2

Requirement Upon detection of a fire adjacent to a loaded HI-STORM UMAX VVM, the ISFSI owner shall take the appropriate immediate actions necessary to extinguish the fire. Following the termination of the fire, a visual and radiological inspection of the equipment shall be performed.

Observation: The NRC inspectors reviewed the licensee's prefire plan and associated strategies and fire response procedure. The inspectors found that the immediate response actions by the fire brigade were incorporated into the prefire plan for the ISFSI pad. If a fire occurred on the heavy haul path, procedure HPP-2253-400, Attachment 8.10 required a continuous attendant equipped with a fire extinguisher who could respond. If that individual could not handle the fire, the fire brigade would be available to respond and have been trained on possible ISFSI transport fires since the ISFSI heavy haul path and pad were entirely within the reactor protected area.

Procedure FPP-ZZ-00007 required a visual and radiological inspection after a fire. There were additional requirements in the Holtec FSAR for actions to take after a fire that were not documented in a site procedure. This observation was captured in CAR 201501626 and the licensee performed a revision to OTO-KC-00001 that incorporated the required actions discussed in the FSAR. The inspectors reviewed the revised procedure and verified the FSAR requirements were adequately incorporated.

The inspectors also reviewed the licensee's EALs and determined that the site did classify a fire associated with the ISFSI causing confinement boundary damage as an Unusual Event.

Documents Reviewed: a) Callaway Procedure FPP-ZZ-00007, "Prefire Strategies," Revision 15; b) Holtec Procedure HPP-2253-400, "MPC Transfer at Callaway," Revision 7; c) Callaway Procedure EIP-ZZ-00101 Addendum 1, "Emergency Action Level Classification Matrix," Revision 4; d) Callaway Procedure OTO-KC-00001, "Fire Response," Revision 14

Category: Fire Protection **Topic:** Fire and Explosion Hazards Analysis

Reference: CoC 1040, App. B.3.4.5; FSAR 1040, Sect 2.2.3.3 Amendment 0/Revision 2

Requirement The potential for fire or explosion shall be addressed, based on site specific considerations. This includes the condition that the onsite transporter fuel tank will contain no more than 50-gallons of diesel fuel while handling a loaded storage cask or transfer cask.

Observation: Explosion hazards were analyzed along the haul path and near the ISFSI in Holtec Report HI-2146196. They included hydrogen tanks and lines, gasoline tanks, and hydrogen delivery trucks since they were adequately determined to be the credible explosion hazards. The assumptions used for explosive hazards in the fire hazards analysis appeared reasonable. No credible explosion hazard exceeded the overpressure needed to tip over the HI-TRAC VW during transport operations or the structural limits of the closure lid for the HI-STORM UMAX at the ISFSI. The fire hazards analysis determined the minimum distance from the ISFSI pad and maximum amount for the hydrogen deliveries that the site had incorporated into procedures to ensure the explosion analysis remained bounding. Since the hydrogen deliveries were not analyzed during transfer using the HI-TRAC VW, the licensee incorporated procedural steps in HPP-2253-400 to stop all hydrogen deliveries during transport operations.

Fire hazards were also analyzed along the haul path and near the ISFSI in Holtec Report HI-2146196. They included diesel tanks and pipes, lube oil tanks and equipment, transformers, gasoline tanks, hydrogen tanks and pipes, delivery trucks of flammable liquids, and Class A flammable materials. The assumptions used for fire hazards in the fire hazards analysis appeared reasonable except that a few buildings near the ISFSI pad did not appear to have been evaluated. These buildings were identified during the NRC inspectors' walk down of the haul path. As a result, the licensee evaluated those buildings and found that they were bounded by other fires in the analysis. No credible fire hazard was found to exceed the acceptable heat input to either the HI-TRAC VW or HI-STORM UMAX closure lid. The fire hazards analysis determined the minimum distance from the ISFSI pad and maximum amount for the flammable liquid deliveries

that the site had incorporated into procedures to ensure the fire analysis remained bounding. Since the flammable liquid deliveries were not analyzed for the HI-TRAC on the haul path, the licensee will stop all flammable liquid deliveries during transport operations. The requirement to ensure no deliveries are made during transport operations was verified to be incorporated into site procedures. It is also important to note that the ISFSI pad area was designated as a combustible free zone in site procedures to ensure the analysis remains bounding.

During the review of the 72.212 report, the inspectors reviewed the licensee's analysis of the worst case fire during transport operations to determine whether it was bounded by the analyzed fire in the HI-STORM UMAX FSAR of 50 gallons of diesel fuel from the cask transporter. A piece of equipment named the HI-PORT was used at Callaway to transport the HI-TRAC out of the fuel handling building. Once out of the building, the VCT engaged the HI-TRAC to transport it to the ISFSI pad. The most limiting scenario was when the HI-PORT and VCT were together while the HI-TRAC was being engaged by the VCT. This operation also included the use of an Articulating Boom Lift next to the HI-PORT and VCT. This combined fire hazard included 50 gallons of diesel fuel and 380 gallons of hydraulic fluid from the VCT; 50 gallons of diesel fuel, 130 gallons of hydraulic fluid, and 48 tires with rubber and polyurethane elastomer from the HI-PORT; and a relatively small amount of hydraulic fluid from the boom lift. This fire loading exceeded that of the HI-STORM UMAX FSAR and was specifically evaluated in Holtec Report HI-2156590, "Evaluation of Combined Effect of HI-PORT and VCT Fires on HI-TRAC at Callaway." The evaluation determined that the fuel temperature, MPC component temperatures, and MPC cavity pressure remained well below their limits and the combined fire event did not exceed any FSAR fire accident acceptance criteria. The inspectors determined that the assumptions in the evaluation were reasonable. The inspectors then reviewed the licensee's controls during this operation that would ensure the analysis would remain bounding but did not find it specifically discussed in the applicable procedures. Due to this observation, the licensee updated the procedures to ensure this operation occurred as evaluated and the inspectors verified the revision change to the procedure was adequate. In addition, the inspectors requested the licensee to provide the 72.48 evaluation, since the site-specific fire hazard exceeded what was analyzed in the FSAR. The licensee provided the 72.48 evaluation, which determined this activity did not require NRC prior review and approval. The inspectors found that the 72.48 evaluation was performed adequately.

The inspectors reviewed the Hazards Walkdown Checklist in procedure HPP-2253-400 and performed a walk down of the haul path to ensure adequate controls were in place to limit combustibles along the haul path and all fire and explosion hazards were analyzed. Only the buildings that could contain Class A combustibles or flammable liquids discussed above were found not to have been analyzed, but were ultimately determined to be bounded by other more limiting fire hazards. The inspectors discussed a number of observations with the licensee to clarify and improve the checklist, which were incorporated into a revision of the checklist and associated procedure. The checklist ensured, among other things, that transient combustibles were controlled to adequate distances from the haul route, all "Hot Work" activities near the haul path were suspended, sufficient fire protection equipment was available along the haul route, and there were no fire protection impairments related to the equipment needed to respond to

fires along the haul route or at the ISFSI pad during transportation operations.

Documents Reviewed: a) Holtec Procedure HPP-2253-400, "MPC Transfer at Callaway," Revisions 2, 4, 5, and 7; b) Holtec Report HI-2146196, "Evaluation of Plant Hazards at Callaway Energy Center," Revisions 2, 3, and 4; c) Holtec Report HI-2156590, "Evaluation of Combined Effect of HI-PORT and VCT Fires on HI-TRAC at Callaway," Revisions 0 and 1; d) Holtec Report HI-2135677, "Evaluation of Effects of Tracked VCT Fire on HI-STORM FW System," Revision 5; e) Holtec Report HI-2094400, "Thermal Evaluation of HI-STORM FW," Revision 12; f) 72.48 Evaluation titled "MP 14-0014, Dry Fuel Storage Licensing and Operations Documentation," Revision 0; g) 10 CFR 72.212 Evaluation Report "Callaway Plant, Unit 1, Dry Fuel Storage System for Spent Nuclear Fuel Docket 72-1045," Revision 0

Category: Fire Protection **Topic:** Fire Protection Plan

Reference: 10 CFR 50.48(a)(1) Published 2015

Requirement Each operating nuclear power plant must have a fire protection plan that satisfies Criterion 3 of Appendix A to Part 50. This fire protection plan must describe the overall fire protection program for the facility.

Observation: Callaway had incorporated the ISFSI into the applicable fire protection procedures and program documents to ensure it was adequately protected. This included 1) control of combustibles in the vicinity of the ISFSI pad, 2) that transient combustibles were evaluated near the ISFSI, 3) site modifications took into account fire hazards to the ISFSI, 4) delivery of combustible and explosive materials was suspended during cask transport operations, and 5) the quantities of combustible and explosive materials delivered during storage operations were within the limits specified in the site specific fire hazards analysis and Holtec FSAR.

Documents Reviewed: a) Callaway Procedure APA-ZZ-00700, "Fire Protection Program," Revision 20; b) Callaway Procedure APA-ZZ-00741, "Control of Combustible Materials," Revision 27

Category: Fuel Selection/Verification **Topic:** Authorized Contents For Storage

Reference: CoC 1040, Appendix B, Section 2.1.1 Amendment 0

Requirement The HI-STORM UMAX system and MPC-37 is authorized for storage of fuel assemblies, fuel debris, and non-fuel hardware meeting the requirements of Appendix B, Section 2.1.1 and Tables 2.1-1 through 2.1-3.

Observation: The licensee had adequately incorporated the authorized contents for storage requirements into their site procedures. For the initial campaign, the licensee planned to limit loading to intact fuel elements only. No damaged fuel would be loaded for dry storage per the acceptance criteria of Section 3.0 of ETP-ZZ-04020. Section 5.4 of the document provided limits for Callaway storage. Page 165 presented limits for comparison with those of Table 2.1-1 through 2.1-3 of the Certificate of Compliance. Other limits in agreement with the CoC were specified in Section 5.1, Precautions and Limitations, which presented limits on MWD/MTU, heat load per cell, cladding type, maximum initial enrichment, post irradiation cooling time, fuel assembly length and width and weight. Section 5.5 of the procedure documented compliance with Note 1 of

Table 2.1-1 of the CoC. The first canister's fuel assemblies information was reviewed by NRC inspectors and was found to meet requirements of Appendix B, Section 2.1.1 and Tables 2.1-1 through 2.1-3.

Documents Reviewed: a) Callaway Procedure ETP-ZZ-04020, "Fuel Selection and Cask Loading for Dry Cask Storage," Revision 0

Category: Fuel Selection/Verification **Topic:** Damaged Fuel Classification

Reference: FSAR 1040, Glossary; ISG-1, Rev. 2 Revision 2

Requirement A damaged fuel assembly is a fuel assembly with known or suspected cladding defects, as determined by review of records, greater than pinhole leaks or hairline cracks, empty fuel rod locations that are not replaced with dummy fuel rods, missing structural components such as grid spacers, whose structural integrity has been impaired such that geometric rearrangement of fuel or gross failure of the cladding is expected based on engineering evaluations, or those that cannot be handled by normal means. Fuel assemblies that cannot be handled by normal means due to fuel cladding damage are considered fuel debris.

Observation: The licensee adequately implemented the damage fuel classification. Damaged fuel was defined in Section 5.4.7 of ETP-ZZ-04020. The section stated that damaged fuel assemblies were those with known or suspected cladding defects, as determined by a review of records, greater than pinhole leaks or hairline cracks, empty fuel rod locations that are not filled with dummy fuel rods, missing structural components such as grid spacers, whose structural integrity had been impaired such that geometric rearrangement of fuel or gross failure of the cladding was expected based on engineering evaluations or that cannot be handled by normal means. Fuel assemblies that cannot be handled by normal means due to fuel failure were considered fuel debris.

Documents Reviewed: a) Callaway Procedure ETP-ZZ-04020, "Fuel Selection and Cask Loading for Dry Cask Storage," Revision 0

Category: Fuel Selection/Verification **Topic:** Decay Heat, Burnup & Cooling Time Limits

Reference: CoC 1040, App. B Table 2.1-1, Table 2.3-1 Amendment 0

Requirement Fuel assemblies stored in the HI-STORM UMAX system canister must meet the decay heat, burnup and cooling time limits specified in Appendix B, Table 2.1-1 and Table 2.3-1 of the Certificate of Compliance.

Observation: Callaway Procedure ETP-ZZ-04020, incorporated the decay heat, burnup, and cooling time requirements from the Certificate of Compliance (Appendix B. 2.1 and 2.3 of CoC). Section 5.1 of the Procedure, outlined precautions and limitations for assemblies to be placed in dry cask storage. It was recognized in the procedure that no damaged fuel is to be loaded at this time. In addition, requirements for assembly burnup and heat load per cell limits agreed with limits established in the CoC. Section 5.4 of the procedure, outlined that only fuel assemblies meeting specific limits are approved for storage. Criteria outlined in Section 5.4 for cladding type (Zr), maximum initial enrichment (5 wt %), post irradiation cooling time (equal to or greater than 3 years), fuel assembly width (equal to or less than 8.54 inches) and fuel assembly weight (equal to or

less than 2050 pounds) are identical to those presented in Table 2.1-1 of the CoC. Acceptance criteria for fuel assembly length of equal to or greater than 114 inches and equal to or less than 168 inches compared with the CoC limit of equal to or less than 199.2 inches. Fuel assembly decay heat values were determined utilizing the Caskloader computer code developed by the Electrical Power Research Institute (EPRI) to assist utilities in the planning, preparation, and execution of loading spent fuel assemblies and core components into dry fuel storage casks. Calculations for regional loading were made by the licensee to comply with Appendix B Section 2.3, Table 2.3-1, utilizing Append B Figure 2.3-1. Non-fuel hardware decay heat values were utilized for particular storage locations.

Documents Reviewed: a) Callaway Procedure ETP-ZZ-04020, "Fuel Selection and Cask Loading for Dry Cask Storage," Revision 0

Category: Fuel Selection/Verification **Topic:** Fuel Loading Error

Reference: CoC 1040, Appendix B, Section 2.2 Amendment 0

Requirement If any loading condition of Appendix B, Section 2.1 is violated, the affected fuel assemblies shall be placed in a safe condition, the NRC Operations Center shall be notified within 24 hrs, and a special report describing the cause of the violation and actions taken to restore compliance and to prevent recurrence shall be submitted to the NRC within 30 days.

Observation: The licensee had requirements in their procedure that noted, if any loading condition is violated, the affected assemblies should be placed in a safe condition and notifications made. Attachment 1 of the APA-ZZ-00520 specified on Item 7, sheet 14 of 24 in sub-item, that the NRC Operation Center will be notified initially within 24 hours if any of the Fuel Specifications or Loading Conditions in Section 2.1 of the HI-STORM UMAX CoC, Appendix B were violated. Sheets 17 and 18 noted that a follow-up notification in the form of a special written report was required within 30 days of initial notification. The report was required to describe the cause of the violation and actions taken to restore compliance and prevent reoccurrence.

Documents Reviewed: a) Callaway Procedure APA-ZZ-00520, "Reporting Requirements and Responsibilities," Revision 44

Category: Fuel Selection/Verification **Topic:** Fuel Shims

Reference: FSAR 1032 Section 1.2.1.1 Revision 3

Requirement The actual length of fuel shims (if required) will be determined on a site-specific and fuel assembly-specific basis.

Observation: Callaway had implemented provisions to ensure adequate shims were installed into the MPCs prior to loading the canisters. Fuel shims were utilized to vertically position fuel assemblies in the canister to ensure the gap between the MPC lid and the top of the fuel insert, or top nozzle of the assembly (with no insert), met the analyzed limits of 1.5 to 2.5 inches. Axial clearance was provided to account for manufacturing tolerances and the irradiation and thermal growth of fuel assemblies. Actual length of the fuel shims were determined based of site-specific and fuel assembly specific bases. Section 7.0 of

the procedure ETP-ZZ-0420 provided the applicable instructions. Step 7.3.2 stated that the licensee must ensure that the specified MPC had been loaded with fuel spacers appropriate for the insert type. Step 7.6.2 noted that the full down position of fuel assemblies will vary based on the insert type (No insert, RCCA, or BP/TP/NSA).

Documents Reviewed: a) Callaway Procedure ETP-ZZ-0420, "Fuel Selection and Cask Loading for Dry Cask Storage," Revision 0

Category: Fuel Selection/Verification **Topic:** Material Balance, Inventory, and Records

Reference: 10 CFR 72.72(a) Published 2015

Requirement Each licensee shall keep records showing the receipt, inventory (including location), disposal, acquisition, and transfer of all SNM with quantities specified in 10 CFR 74.13(a)(1).

Observation: The licensee's Special Nuclear Material (SNM) accountability plan was revised to include fuel transferred from the spent fuel pool to Callaway's ISFSI. Details of the SNM accountability plan were presented in Procedure APA-ZZ-00405. A new Item Control Area was created for the ISFSI in Revision 28 of the procedure.

Documents Reviewed: a) Callaway Procedure APA-ZZ-00405, "Special Nuclear Material Control and Accounting Procedure," Revision 28

Category: Fuel Selection/Verification **Topic:** Post Loading Verification

Reference: FSAR 1032, Section 9.2.3.3 Revision 3

Requirement Perform a post-loading visual verification of the assembly identification markings to confirm that the serial numbers match the approved fuel loading pattern.

Observation: The licensee's procedures had incorporated the requirement to perform an independent post-loading visual verification to confirm that the fuel assembly serial numbers matched the loading plan. The post verification requirement was specified in step 7.10 of ETP-ZZ-04020. NRC inspectors observed the licensee perform the post loading verification, utilizing an underwater camera on the first canister loaded during the week of August 24, 2015.

Documents Reviewed: a) Callaway Procedure ETP-ZZ-04020, "Fuel Selection and Cask Loading For Dry Cask Storage," Revision 0

Category: General License **Topic:** Evaluation of Effluents/Direct Radiation

Reference: 10 CFR 72.212(b)(5)(iii) & 10 CFR 72.104(a) Published 2015

Requirement The general licensee shall perform a written evaluation prior to use that establishes that the requirements of 10 CFR 72.104 "Criteria for Radioactive Materials in Effluents and Direct Radiation from an ISFSI" have been met. 10 CFR 72.104 requires the annual dose equivalent to any real individual who is located beyond the controlled area must not exceed 25 mrem to the whole body, 75 mrem to the thyroid and 25 mrem to any other critical organ during normal operations and anticipated occurrences,

Observation: The licensee performed an evaluation to ensure that the requirements of 10 CFR 72.104,

Criteria for Radioactive Materials in Effluents and Direct Radiation from an ISFSI and showed in Section 5.3 of the 72.212 Report, that the dose equivalent to any individual who is located outside the controlled area was a small fraction of the regulatory limits of 25 mrem to the whole body, 75 mrem to the thyroid, and 25 mrem to any other critical organ. No effluent doses, including thyroid doses, would occur with the welded and sealed Holtec canisters used by the Callaway Energy Center. Though there was not a berm on all sides of the ISFSI, for flood control purposes there were berms on two sides, plant south and plant east that contribute to shielding of direct radiation. Further discussion of calculations associated with the doses to the public are provided under the Category: Radiation Protection and the Topic: Controlled Area Boundary Dose Rate Analysis.

Documents Reviewed: a) 10 CFR 72.212 Evaluation Report, "Callaway Plant, Unit 1, Dry Fuel Storage System for Spent Nuclear Fuel Docket 72-1045," Revision 0; b) Callaway Procedure HPCI-15-05, "Evaluation of Direct Radiation Dose to the Member of the Public from the Independent Spent Fuel Storage Installation," Revision 1

Category: General License **Topic:** Flood Conditions

Reference: CoC 1040, Appendix B, Section 3.4.4 Amendment 0

Requirement Verify that the site analysis for flooding does not exceed the Certificate of Compliance limits of 15 fps water velocity and a height of 125 feet of water.

Observation: The UMAX at the Callaway site was evaluated and verified to not exceed the Certificate of Compliance limits for 15 fps velocity of water and a height of 125 feet of water. Section 5.4.1.3 of the 72.212 Evaluation Report documented that all of Callaway's plant safety-related structures and components are located on a plateau 280 feet or more above maximum flood level. The site is higher than the surrounding terrain so natural streams drain away from the plateau so that isolated local flooding will not occur as a result of a probable maximum precipitation event. The GEI calculation CEC-CS 006-01, concluded that the peak water surface elevations resulting from local intense precipitation event does not exceed critical elevations for the Callaway ISFSI site. Section 5.4.1.7 of the 72.212 Evaluation Report, noted that the ISFSI site is located on a slight plateau with no significantly higher ground within five miles of the site.

The HI-STORM UMAX System is flood resistant as discussed in Section 2.4.7 of the UMAX FSAR. Licensee documents noted that the VVM will withstand a hydraulic head of 125 feet of water submergence. Full or partial submergence of the MPC is not of concern as heat removal is enhanced by the presence of water.

Documents Reviewed: a) 10 CFR 72.212 Evaluation Report, "Callaway Plant, Unit 1, Dry Fuel Storage System for Spent Nuclear Fuel Docket 72-1045," Revision 0; b) Consultant Calculation GEI, CEC-CS 006-01, "Callaway Site Evaluation," Revision 0

Category: General License **Topic:** Initial Compliance Evaluation Against CoC

Reference: 10 CFR 72.212(b)(5) Published 2015

Requirement A general licensee shall perform written evaluations, prior to use and before applying the changes authorized by an amended Certificate of Compliance to a cask loaded under the

initial Certificate of Compliance or an earlier amended Certificate of Compliance, which establishes that the cask, once loaded with spent fuel or once the changes authorized by an amended Certificate of Compliance have been applied, will conform to the terms, conditions, and specifications of the Certificate of Compliance or amended Certificate of Compliance listed in 10 CFR 72.214.

Observation: The Callaway 72.212 Evaluation Report evaluated the terms, conditions and specifications in Certificate of Compliance 1040, Amendment 0, and documented that the conditions as set forth had been met at the Callaway site. Section 5.1 of the 72.212 Evaluation Report, provided a detailed comparison of the requirements in the Certificate of Compliance against the procedures and programs established at Callaway. The licensee used a combination of already existing Part 50 programs and procedures and newly developed procedures specifically for the ISFSI.

Vertical Ventilated Modules (VVMs) having a cavity enclosure container (CEC), a cavity enclosure divider shell, a VVM Closure lid, the HI-TRAC VW transfer cask, and the MPC-37 were to be used, at Callaway, to transport and store the pressurized water reactor spent fuel. The MPC-37 canister holds 37 spent fuel assemblies.

Appendix 1 of the 72.212 Evaluation Report provided a detailed list of each of the licensing conditions in the Certificate of Compliance, the technical specifications in Appendix A of the CoC, and the approved contents and design requirements in Appendix B of the CoC and how Callaway complied with these licensing requirements.

License Condition 1 required the licensee to develop written operating procedures for cask handling, loading, movement, surveillance, and maintenance that are consistent with the UMAX FSAR description of these activities. The licensee had developed "pool-to-pad" and other operating procedures that cover all the activities in License Condition 1.

License Condition 2 required that written cask acceptance tests and a maintenance program be established which is consistent with the technical basis in the UMAX FSAR. Compliance with License Condition 2 was demonstrated through Holtec Component Completion Records and CoC's issued for important to safety structures, systems, and components. The required procedures involving Acceptance Criteria and the Maintenance Program for Callaway dry fuel storage operations had been prepared, revised, and issued as necessary.

License Condition 3 required that activities being performed related to structures, systems, and components designated as important to safety be conducted under an NRC approved quality assurance program. Callaway was using their Part 50 approved program for Part 72 activities.

License Condition 4 required that each lift of an MPC or a HI-TRAC VW transfer cask be performed in accordance with existing heavy load requirements and procedures of the licensed facility. Lifts within the Callaway Fuel Building and made by the single failure proof crane were governed by 10 CFR Part 50. Dry Fuel Storage lifts were conducted in accordance with existing heavy loads Procedure APA-ZZ-00365 and other new Dry Fuel Storage heavy load handling procedures which met existing heavy load requirements.

License Condition 5 required that the fuel loaded in the MPC-37 meet the required fuel specification in Appendix B of the Certificate of Compliance 1040. During the NRC inspection, a detailed review of the licensee's program related to fuel selection was completed to verify that processes and procedures had been put in place by the licensee to ensure that only spent fuel consistent with Appendix B of CoC 1040 would be selected for storage. More detail related to selection of the spent fuel is found in the Inspector Notes of this report under the Category "Fuel Selection and Verification."

License Condition 6 required that features and characteristics for the site or system must be in accordance with Appendix B of the CoC. A discussion of each of the design features in Appendix B were discussed in Appendix 1, Table 3 of the 72.212 Evaluation Report. References were often provided to various Callaway procedures and documents to show compliance with the design features listed in Appendix B.

License Condition 7 discussed making changes to the Certificate of Compliance. Only Holtec can make requests to the NRC to change CoC 1040, since they were the certificate holder. Callaway was a general licensee, and therefore cannot make any change requests directly to the NRC.

License Condition 8 lists the required pre-operational testing and training exercises for the UMAX system prior to the first actual use of the storage system. Callaway had completed all the required testing and training exercises. Details related to each of these can be found in the Inspector Notes of this inspection report under the Category "Dry Run Demonstration."

License Condition 9 authorized use of the Holtec HI-STORM UMAX Canister Storage System as a general license if the user possessed a Part 50 license. Callaway will use the UMAX system as a general license licensee and currently holds a Part 50 reactor license for the single unit Callaway Plant under Docket Number 50-483.

The 72.212 Evaluation Report, Appendix 1, Table 2, CoC Appendix A – Technical Specifications evaluated the requirements of the Callaway dry fuel storage program against the technical specifications in Appendix A of the Holtec 1040 license. Each of the technical specification requirements in the CoC, Appendix A were evaluated. Table 2 addressed and discussed each of the technical specification requirements and when required, referenced where the requirement had been incorporated into the Callaway procedures.

The requirements in the CoC, Appendix B for the approved content and design features were compared to the Callaway dry fuel storage program and documented in the 72.212 Evaluation Report, Appendix 1, Table 3, CoC Appendix B – Approved Contents and Design Features. Each of the applicable requirements from Appendix B of the CoC were shown to be incorporated into Callaway procedures or, for site conditions that were required to be met, Callaway had performed the necessary analysis to show compliance and referenced the analysis document in the table. Many of the table sections also referenced back to specific sections of the main body of the 72.212 Evaluation Report.

Documents Reviewed: a) Callaway Plant Updated Final Safety Analysis Report (UFSAR), The Callaway UFSAR has two parts, the UFSAR-SP for the Standard Plant and the UFSAR-SA for the Site Addendum, Dated February 2014; b) 10 CFR 72.212 Evaluation Report, "Callaway Plant, Unit 1, Dry Fuel Storage for Spent Nuclear Fuel," Revision 0; c) Certificate of Compliance 1040, "Holtec International HI-STORM UMAX Canister Storage System," Amendment 0; d) Holtec Report No. HI-2115090, "Final Safety Analysis Report for the HI-STORM UMAX Canister Storage System," Revision 2; e) Callaway Letter to NRC UNLNRC-06180 Subject: Notification Pursuant to 10 CFR 72.140(d) of Intent to Apply Previously Approved Quality Assurance Program to Independent Spent Fuel Storage Installation (ISFSI) at Callaway Plant, Unit 1, dated February 13, 2015

Category: General License **Topic:** Initial Compliance Evaluation Against FSAR

Reference: 10 CFR 72.212(b)(6) Published 2015

Requirement The general licensee shall review the FSAR referenced in the Certificate of Compliance or amended Certificate of Compliance and the related NRC Safety Evaluation Report, prior to use of the general license, to determine whether or not the reactor site parameters, including analysis of earthquake intensity and tornado missiles, are enveloped by the cask design basis considered in these reports. The results of this review must be documented in the evaluation made in 10 CFR 72.212(b)(5).

Observation: The licensee documented the required written evaluations in the 72.212 Evaluation Report as Section 5.4, "10 CFR 72.212(b)(6) Reactor Site Parameters Review of UMAX FSAR and SER." Section 5.4 included site specific analysis of fires and explosions, tornados, floods, hurricanes and extreme winds, earthquakes, lightning, burial under debris, environmental temperatures, and snow/ice. Each topical area was reviewed against the Callaway Updated Final Safety Analysis Report (UFSAR) or other site specific documents. The licensee found that they were bounded by the UMAX FSAR in all areas except for fire, explosions, and tornados. As such, Callaway performed a 72.48 evaluation and additional calculations to show the design basis for each area was still met and did not require Callaway to have the Certificate holder to request a license amendment (See Topic Safety Reviews).

Section 5.2.2 of the 72.212 report documented the review of the ISFSI stability including seismic evaluations. The ISFSI stability was evaluated by examining the liquefaction potential, settlement, bearing capacity, and potential for sliding and overturning. The design base earthquake event for the Callaway site was evaluated to not create an issue with MPC retreivability, subcriticality, and the confinement boundary would not be compromised. Callaway Part 50 UFSAR Figure 3.7(B)-1 and Figure 3.7(B)-2 described the design basis earthquake of the Callaway site as a horizontal and vertical zero period accelerations (ZPAs) of 0.20g for both directions. The ground surface horizontal and vertical ZPA used to seismically analyze the UMAX structure in Section 3.4.4.1.2 of the HI-STORM UMAX FSAR ranged from values of 0.539g to 1.008g, with peak accelerations up to 4.040g. Callaway's design basis earthquake was bounded by the UMAX design basis by a considerable margin.

NRC inspectors reviewed the Callaway's 72.212 evaluation report and supplemental evaluations which demonstrated the site parameters for all environmental conditions

analyzed were enveloped by the HI-STORM UMAX system design bases. No findings or issues were identified during this review.

Documents Reviewed: a) 72.48 Evaluation Log No. 15-01, "MP 14-0014, Dry Fuel Storage Licensing and Operations Documentation," Revision 0; b) Holtec Report HI-2146196, "Evaluation of Plant Hazards at Callaway Energy Center," Revisions 2, 3, and 4; c) Holtec Report HI-2156590, "Evaluation of Combined Effect of HI-PORT and VCT Fires on HI-TRAC at Callaway," Revisions 0 and 1; d) Holtec Report HI-2135677, "Evaluation of Effects of Tracked VCT Fire on HI-STORM FW System," Revision 5; e) 10 CFR 72.212 Evaluation Report, "Callaway Plant, Unit 1, Dry Fuel Storage System for Spent Nuclear Fuel Docket 72-1045," Revision 0

Category: General License **Topic:** Initial Evaluation Against Part 50 License

Reference: 10 CFR 72.212(b)(8) Published 2015

Requirement Prior to use of the general license, determine whether activities related to storage of spent fuel involve a change in the facility technical specifications or require a license amendment for the facility pursuant to 10 CFR 50.59(c). Results of this determination must be documented in the evaluation made in 10 CFR 72.212(b)(8).

Observation: Activities related to the storage of spent fuel under the Callaway 10 CFR Part 72 general license were reviewed in accordance with 10 CFR 50.59 for the required modifications and procedure changes, and none of these reviews resulted in NRC approval being required. Callaway developed Modification Package MP14-0014, Dry Fuel Storage Licensing and Operations Documentation to assess and authorize the dry fuel storage operations at Callaway Unit 1. The purpose of MP 14-0014 was to address all operations for loading spent fuel assemblies into an MPC, drying and sealing of the MPC, transport of the loaded transfer cask to the Callaway UMAX ISFSI, transfer of the loaded MPC from the transfer cask into the UMAX, storage of the spent fuel in the UMAX system at Callaway, and all unloading operations. Other modification packages required for ISFSI implementation, including Callaway ISFSI installation, haul path construction, security modifications, and upgrade of the cask handling crane in the fuel building were located in different modification packages that were listed in Table 5.5 of Callaway 72.212 Evaluation Report. NRC inspectors reviewed many of listed modification packages throughout the construction of the ISFSI, dry run inspections, and first loading inspection. No findings were identified regarding Callaway's 50.59 reviews.

Documents Reviewed: a) 10 CFR 72.212 Evaluation Report, "Callaway Plant, Unit 1, Dry Fuel Storage for Spent Nuclear Fuel," Revision 0; b) 50.59 Screen MP14-0014, "Dry Fuel Storage Licensing and Operations Documentation," Revision 0

Category: General License **Topic:** ISFSI Decommissioning Funding Report

Reference: 10 CFR 72.30 (b) Published 2015

Requirement Each licensee must submit for NRC review and approval a decommissioning funding plan prior to loading fuel into an ISFSI.

Observation: Callaway provided their draft letter to the NRC inspectors for the ISFSI Decommissioning Funding Report to be submitted to the NRC before their initial loading

campaign. CAR 201306666 was tracking this submission with a due date before the initial load commenced. The funding plan letter was ultimately submitted prior to the first loading campaign to the NRC on August 17, 2015 (ML15229A127).

Documents Reviewed: a) Letter from Callaway to NRC, "Docket Numbers 50-483 and 72-1045 Callaway Plant Unit 1 Union Electric Co. Renewed Facility Operating License NPF-30, ISFSI Decommissioning Funding Plan," Dated August 17, 2015 (ML15229A127); b) Draft 10 CFR 72.30 ISFSI Decommissioning Cost Estimate

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Category: General License **Topic:** Limiting Site Temperatures

Reference: CoC 1040, Appendix B, Section 3.4.1; 3.4.2 Amendment 0

Requirement The maximum average yearly temperature at the site shall be verified as 80 degrees F. The temperature extremes, averaged over a 3-day period, shall be greater than -40 degrees F and less than 125 degrees F.

Observation: The Callaway site is below the maximum average yearly temperature of 80 degree F and within the -40 to 125 degree F temperature extremes, averaged over any 3-day period. The Callaway UFSAR-SA (Site Addendum), Section 2.3.2.1.2.2, "On-Site Temperatures," stated that the annual average temperature in the site area was approximately 55.6°F, and was therefore enveloped by the HI-STORM UMAX Canister Storage System design basis. Table 2.3.6 of the HI-STORM UMAX FSAR specified a design basis normal soil temperature (bounding annual average) not to exceed 77°F. Chapter 11, Page 11.10 of the 2008 ASHRAE Handbook - HVAC Systems and Equipment (Reference f), stated that the average annual air temperature should be used to approximate the average annual soil temperature. As stated above, the annual average air temperature at the Callaway site was approximately 55.6°F, which was less than the 77°F requirement specified in the HI-STORM UMAX FSAR for annual average soil temperature.

The lower bound off-normal temperature limit for the HI-STORM UMAX in the HI-STORM UMAX FSAR, Table 2.3.6, was defined as a 3-day average minimum ambient temperature of -40°F. Callaway UFSAR-SA, Table 2.3-23, showed a lowest extreme minimum temperature of -26°F. Therefore, the Callaway minimum temperature met the -40°F 3-day average minimum design basis temperature limit in the HI-STORM UMAX FSAR.

Table 2.3-22 of the Callaway UFSAR-SA showed 116°F as the highest extreme maximum temperature. This was bounded by the extreme accident level ambient temperature limit of 125°F for which the HI-STORM UMAX VVM was designed.

Documents Reviewed: a) Holtec Report No. HI-2115090, "Final Safety Analysis Report on the HI-STORM UMAX Canister Storage System (HI-STORM UMAX FSAR), Docket 72-1040," Revision 2; b) Callaway Plant Updated Final Safety Analysis Report (UFSAR), The Callaway UFSAR has two parts, the UFSAR-SP for the Standard Plant and the UFSAR-SA for the Site Addendum, dated February 2014; c) Holtec Procedure, HPP-2253-400 "MPC Transfer at Callaway," Revision 2; d) Holtec Procedure HPP-2253-500, "MPC Unloading at Callaway," Revision 1; e) Certificate of Compliance 1040 HI-STORM UMAX Appendix B Section 3.4.1 and 3.4.2, Amendment 0; f) ASHRAE Handbook,

"HVAC Systems and Equipment," 2008 Edition

Category: General License **Topic:** Program Review - RP, EP, QA, and Training

Reference: 10 CFR 72.212(b)(10) Published 2015

Requirement The general licensee shall review the reactor emergency plan, quality assurance program, training program, and radiation protection program to determine if their effectiveness is decreased and, if so, prepare the necessary changes and seek and obtain the necessary approvals.

Observation: The Callaway Part 50 emergency plan, quality assurance program, training program, and radiation protection program were evaluated for any potential decreases in their effectiveness associated with the implementation of the dry fuel storage operations and equipment. Callaway had revised the Callaway Part 50 Radiological Emergency Response Plan consistent with NEI-99-01, Methodology for Developing Emergency Action Levels. A new ISFSI Emergency Action Level had been incorporated into their emergency program and implementing procedures. Callaway had revised their 10 CFR 50 Appendix B Operating Quality Assurance Manual to include dry fuel storage activities in accordance with 10 CFR 72. Callaway had applied their Training Program, the associated appendices, and training manuals to dry fuel storage activities. Callaway training program relied on Holtec's approved training program for training on pool-to-pad procedures and operations. The remaining training areas, such as, fuel handling, 72.48 reviews, ISFSI quality assurance, ISFSI emergency plan, ISFSI routine Technical Specification surveillances, and ISFSI maintenance of important-to-safety equipment would be implemented under the Callaway training program.

Callaway reviewed and revised the Callaway Radiation Protection Program and implementing procedures to verify it applies to and adequately controls ISFSI and dry fuel storage activities. Existing procedures had been reviewed and revised to ensure posting requirements, access control, survey requirements, and personnel and environmental monitoring are applied to ISFSI activities consistent with 10 CFR Part 20 and Part 72 requirements. Additionally, numerous new ISFSI radiation protection procedures were developed to implement specific dry fuel storage radiological requirements.

Documents Reviewed: a) 10 CFR 72.212 Evaluation Report, "Callaway Plant, Unit 1, Dry Fuel Storage for Spent Nuclear Fuel," Revision 0; b) Callaway Plant 10 CFR 50.47 Radiological Emergency Response Plan, Revision 46; c) Callaway 10 CFR 50 Appendix B Operating Quality Assurance Manual (OQAM), Revision 31; d) Callaway Procedure APA-ZZ-00925, "Training and Qualification of Plant Personnel," dated 2015

Category: General License **Topic:** Revisions to 72.212 Analysis

Reference: 10 CFR 72.212(b)(7) Published 2015

Requirement The general licensee shall evaluate any changes to the written evaluations required by 10 CFR 72.212(b)(5) and 10 CFR 72.212(b)(6) using the requirements of 10 CFR 72.48(c). A copy of this record shall be retained until spent fuel is no longer stored under the general license issued under 10 CFR 72.210.

Observation: The 10 CFR 72.212 Evaluation Report, Section 4.0 stated that Callaway's 72.212 report would be reviewed, and revised if necessary, at a minimum before each dry fuel storage campaign to identify the applicable CoC amendment, FSAR revision, and any changes to the MPC model, fuel loading requirements, or VVM design changes to be used at the ISFSI. Additionally, the changes would be handled in accordance with 10 CFR 72.212(b)(7) to perform an written evaluation through the 72.48 process.

Documents Reviewed: a) 10 CFR 72.212 Evaluation Report, "Callaway Plant, Unit 1, Dry Fuel Storage System for Spent Nuclear Fuel Docket 72-1045," Revision 0

Category: General License **Topic:** Storage Cask Blocked Inlet or Outlet Air Vents

Reference: CoC 1040, Appendix B, Section 3.4.9 Amendment 0

Requirement For those users whose site specific design basis includes an event that results in blockage of the storage cask inlet or outlet air vents for an extended period of time longer than the completion time in LCO 3.1.2 (i.e. 64 hours for heat loads less than or equal to 28.74 kW and 24 hours for heat loads greater than 28.74 kW), an analysis may be performed to demonstrate adequate heat removal for the duration of the event. If the analysis is not performed or adequate heat removal cannot be verified, alternate methods of cooling must be established.

Observation: The Callaway ISFSI pad had been designed and located such that flooding and burial due to debris would not occur. The 72.212 Evaluation Report, Section 5.4.1.3 "Flooding" and Section 5.4.1.7 "Burial under Debris" discussed the features of the ISFSI pad that made these events very unlikely. The ISFSI pad was located on nearly a level plateau with elevations ranging from 830 to 850 feet. The elevation of the flood plain of the Missouri River nearest the site is about 525 feet with a Probable Maximum Flood level at elevation 559 feet. Flooding of the Missouri River would never reach the ISFSI. The elevation of the Callaway site is higher than the surrounding terrain and since well-developed natural streams drain the plateau, isolated local flooding would not occur on the Callaway site. Even when due to a severe event such as the Probable Maximum Precipitation.

The Callaway ISFSI is not located in the vicinity of unstable slopes. Section 2.7, Landslides, in Calculation HI-2146196, Evaluation of Plant Hazards at Callaway Energy Center, noted that Section 2.3.2.2.1 of the Callaway FSAR-SA, described the Callaway Plant and ISFSI site as being located on a slight plateau with no significantly higher ground within 5 miles of the site. Therefore, the site topography precluded the possibility of a landslide hazard that could collapse and surround a HI-STORM UMAX VVM. Calculation HI-2146196 Section 2.2, Fall Hazards, stated that there were no structures in the vicinity of the Callaway ISFSI that could collapse and surround a HI-STORM UMAX VVM. The controlled area around the Callaway ISFSI precluded the close proximity of substantial amounts of vegetation. Therefore, a burial under debris accident affecting the Callaway ISFSI VVMs was not an anticipated event.

For the Callaway ISFSI, HI-STORM UMAX CoC Appendix B, Approved Contents and Design Features Subsection 3.4.12, required performance of an analysis or evaluation if the site-specific design basis includes an event that resulted in the blockage of any HI-STORM UMAX inlet or outlet air duct for an extended period of time and provisions for

alternate means of cooling be established if the fuel cladding short term temperature cannot be demonstrated to be met or if the analysis or evaluation was not performed.

As described above, flooding was not an issue for the Callaway ISFSI, and the 72.212 report concluded that the burial-under-debris accident event does not affect the safe operation of the HI-STORM UMAX Canister Storage System, if the blockage was removed in the Appendix A Technical Specification 3.1.2 specified time period. Callaway Procedure OSP-ZZ-00001 implemented this Technical Specification requirement. Therefore, Subsection 3.4.12 did not apply to the Callaway ISFSI.

Documents Reviewed: a) Callaway Plant Updated Final Safety Analysis Report (UFSAR), The Callaway FSAR has two parts, the UFSAR-SP for the Standard Plant and the UFSAR-SA for the Site Addendum, Dated February 2014; b) Holtec Calculation HI-2146196, "Evaluation of Plant Hazards at Callaway Energy Center," Revision 4; c) Certificate of Compliance 1040 HI-STORM UMAX Appendix B Section 3.4.12, Amendment 0; d) 10 CFR 72.212 Evaluation Report, "Callaway Plant, Unit 1, Dry Fuel Storage System for Spent Nuclear Fuel Docket 72-1045," Revision 0

Category: Heavy Loads **Topic:** Closure Lid Lift Lugs Inspection

Reference: FSAR 1040 Section 10.4.1 Revision 2

Requirement Prior to each MPC loading, closure lid lift lugs examination shall be inspect for indications of overstress such as cracks, deformation, wear marks, corrosion, ect.

Observation: The Holtec CoC 1040 FSAR requirement for VVM closure lid lifting lug inspection was placed into Callaway's Procedure ETP-ZZ-04021 in Step 7.1.20. This requirement was adequately placed into Callaway's procedure to be completed prior to each loading of an MPC.

Documents Reviewed: a) Callaway Procedure ETP-ZZ-04021, "ISFSI Loading Campaign Performance – IPTE," Revision 1

Category: Heavy Loads **Topic:** Licensed Facility Heavy Loads Requirements

Reference: CoC 1040, License Condition 4 Amendment 0

Requirement Each lift of a canister, transfer cask, or storage cask must be made in accordance with the existing heavy loads requirements and procedures of the licensed facility at which the lift is made. A plant specific review (under 50.59 or 72.48, if applicable) is required to show operational compliance with existing plant specific heavy loads requirements.

Observation: The heavy lifts and crane operations associated with the dry cask storage operations were performed in accordance with the plant's maintenance department procedures for all heavy lifting activities in and outside of the plant. The licensee's 50.59 screen that was performed on use of the Holtec equipment and the ISFSI operations documented that reviews and evaluations were performed to ensure all activities were in compliance with the Callaway Lifting and Rigging Program.

The licensee's crane was rated to 125-tons. The maximum lift weight during the dry cask storage operations was calculated at approximately 123.5 tons. The licensee

performed a 50.59 evaluation to determine if the seismic and structural analyses performed by Holtec met the Callaway Part 50 Updated Final Safety Analysis Report (UFSAR) requirements for the plant. The conclusion of the 50.59 evaluation determined that the analysis performed by Holtec was within the bounds of the Part 50 UFSAR and utilized the site's approved methodologies.

Documents Reviewed: a) 50.59 Screen MP 14-004, "Dry Fuel Storage Licensing and Operations Documentation (Heavy Loads Review)," Revision 0

Category: Heavy Loads **Topic:** Procedures

Reference: NUREG 0612, Section 5.1.1 (2) Published July 1980

Requirement Procedures should be developed to cover load handling operations for heavy loads that are or could be handled over or in proximity to irradiated fuel or safe shutdown equipment. The procedures should include: a) identification of the required equipment; b) inspections and acceptance criteria required before movement of the load; c) the steps and proper sequence to be followed in handling the load; d) defining the safe load path; and e) special precautions.

Observation: The procedures used at Callaway that embody the requirements of NUREG 0612 for a safe load path were implemented into Callaway's procedures for lifting operations, operation of the cask handling crane, and MPC loading. Both Callaway and Holtec had developed procedures that covered load handling operations for heavy loads in close proximity to irradiated fuel and reactor safe shutdown equipment. The procedures included identifying the safe load path, special precautions, and the identification of required equipment, inspection requirements, and proper rigging sequence to observe when handling a load.

Documents Reviewed: a) Callaway Procedure APA-ZZ-00365, Addendum L, "Callaway Lifting Operations," Revision 18 b) Callaway Procedure OTS-KE-00016, "Operation of the Cask Handling Crane," Revision 22; c) Holtec Procedure HPP-2253-200, "MPC Loading at Callaway," Revision 9

Category: Heavy Loads **Topic:** Safe Load Paths

Reference: NUREG 0612, Section 5.1.1 (1) Published July 1980

Requirement Safe load paths should be defined for the movement of heavy loads to minimize the potential for heavy loads, if dropped, to impact irradiated fuel in the reactor vessel and in the spent fuel pool, or to impact safe shutdown equipment. The path should follow, to the extent practical, structural floor members, beams, etc., such that if the load is dropped, the structure is more likely to withstand the impact.

Observation: Callaway had satisfied the requirement that a safe load path be established as required by NRC NUREG 0612, Section 5.1.1 (1). Callaway's cask handling crane procedure, OTS-KE-00016, Attachment 1, clearly delineated the safe load and travel path for the cask handling crane that would apply when moving the transfer cask. The allowed load path showed exclusion areas for both the main hoist and the auxiliary hoist. The load path protects critical safety systems from being impacted in the event of a load drop scenario.

Documents Reviewed: a) Callaway Procedure OTS-KE-00016, "Operations of the Cask Handling Crane," Revision 22

Category: Heavy Loads **Topic:** Site Temperature Limit for Cask Handling

Reference: CoC 1040, Appendix B, Section 3.4.11 Amendment 0

Requirement Loading, transport, and unloading operations shall only be conducted with working area ambient temperatures of 0 degrees F or higher.

Observation: Holtec's MPC transfer procedure required that the ambient temperature for VCT operation be between 0 and 120 degrees F. This matched the requirements of CoC 1040, Appendix B, Section 3.4.11, which required that loading, transport, and unloading operations shall only be conducted when the work area ambient temperature is 0 degrees F or higher.

Documents Reviewed: a) Holtec Procedure HPP-2253-400, "MPC Transfer at Callaway," Revision 6; b) Holtec Procedure HPP-2253-500, "MPC Unloading at Callaway," Revision 5

Category: Heavy Loads **Topic:** Transporter Initial Acceptance Testing

Reference: NUREG 0554, Section 8.2 Published May 1979

Requirement After the 125% static load test, the crane should be given a full performance test with 100% of the maximum critical load attached, for all speeds and motions for which the system is designed. This should include verifying all limiting and safety control devices.

Observation: NRC inspectors reviewed purchase specifications and accepting testing documentation to verify completion of the transporter load testing requirements. According to documents reviewed by NRC, after the 125% static load test was performed, the VCT was given a performance test with a load slightly over its 100% rated capacity, 415,000 lbs. The 100% performance test included verifying the operation of the lift booms, crawler, emergency stop function, measurement of general clearances, VCT travel, redundant drop protection, and lowering of the load using the emergency hydraulic pump. The test procedure and results met all applicable safety requirements for heavy loads and single failure proof lifting devices.

Documents Reviewed: a) Holtec Procedure HSP-199, "VCT Factory Acceptance Test Procedure," dated April 9-10, 2015

Category: Heavy Loads **Topic:** Transporter Initial Acceptance Testing

Reference: NUREG 0554, Section 8.2 Published May 1979

Requirement The VCT should be static load tested at 125 percent of the maximum critical load. The test should be conducted at all positions generating maximum strain in the bridge and trolley structures and other positions as recommended by the designer or manufacturer.

Observation: The VCT delivered to Ameren for use in their fuel loading campaign was static load tested with a weight of 519,630 lbs., just above 125% of the rated load of 415,000 lbs. This load was held for ten minutes at a height of 1 inch. The acceptance test met the applicable requirements.

Documents Reviewed: a) Holtec Procedure HSP-199, "VCT Factory Acceptance Test Procedure," dated April 9 - 10, 2015

Category: Heavy Loads **Topic:** VCT Transportation Evaluation

Reference: CoC 1040, Tech Spec. A.5.2 Amendment 0

Requirement Between the fuel building and the ISFSI pad, The Transfer Cask, when loaded with spent fuel, may be lifted to and carried at any height necessary during transport operations and MPC transfer, provided that a) The metal body and any vertical columns of the lifting equipment are designed to comply with stress limits of ASME Section III, Subsection NF, Class 3 for linear structures and all vertical compression loaded primary members satisfy the buckling criteria of ASME Section III, Subsection NF; b) the horizontal cross beam and any lifting attachments used to connect the load to the lifting equipment were designed, fabricated, operated, tested, inspected, and maintained in accordance with applicable sections and guidance of NUREG-0612, Section 5.1 and ANSI N14.6; and c) the lifting equipment has redundant drop protection features which prevent uncontrolled lowering of the load.

Observation: The Vertical Cask Transporter (VCT) steel load bearing members were designed to comply with the stress limits of ASME Section III, Subsection NF, Class 3 for linear structures, including vertical compression loaded primary members, as specified in Holtec PS-1120, the VCT purchase specification. PS-1120 required that the VCT be designed for both static and dead loads and dynamic seismic loads and invoked the guidance of NUREG-0612, including Section 5.1.6. PS-1120 required that the cross beams and lifting attachments met the stress limits of ANSI N14.6. The VCT employed redundant drop protection features as specified in the purchase specification. The aforementioned features show that the VCT in use at Callaway met the stress limits and compression buckling requirements of ASME Section III, Subsection NF; the cross beam meets the design requirements of NUREG-0612, and the applicable requirements of ANSI N14.6; and the VCT had redundant drop protection features to prevent uncontrolled lowering of a load.

Documents Reviewed: a) Callaway Dry Storage Project 10 CFR 72.212 Evaluation Report, Revision 0; b) Holtec PS-1120, "Purchase Specification for the Vertical Cask Transporter," Revision 6

Category: Loading Operations **Topic:** Canister Lid Fit Test

Reference: FSAR 1032, Table 10.1.1 Revision 3

Requirement As part of the Holtec inspection and test acceptance criteria, the canister lid, closure ring, and vent and drain port cover plates shall be fit tested prior to canister operation.

Observation: The licensee had incorporated the Holtec FW FSAR requirements into two procedures to perform the required fit tests for the canister lid, closure ring, and vent and drain port cover plates prior to placing the MPC into the spent fuel pool. The licensee utilized Procedures ETP-ZZ-04021 and HPP-2253-100 to perform the fit tests prior to placing the MPC into the spent fuel pool.

Documents Reviewed: a) Callaway Procedure ETP-ZZ-04021, "ISFSI Loading Campaign Performance - IPTE," Revision 1; b) Holtec Procedure HPP-2253-100, "MPC Pre-operational Inspection,"

Revision 3

Category: Loading Operations **Topic:** Cask System Annual Maintenance

Reference: FSAR 1040, Tables 10.4.1 and 10.4.2 Revision 2

Requirement The following cask system maintenance shall be performed annually, or prior to use if out of service for greater than 1 year: a) UMAX VVM in-service inspection, b) ISFSI pad inspection, c) UMAX VVM external surface visual examination, d) UMAX VVM inspection of visual markings, and e) HI-TRAC VW pressure relief valve calibration.

Observation: The licensee had incorporated the UMAX FSAR requirements into three annual Preventive Maintenance procedures (PMs) to perform the required UMAX VVM in-service inspection, ISFSI pad inspection, UMAX VVM external surface visual examination, UMAX VVM inspection of visual markings, and HI-TRAC VW pressure relief valve calibration annually, or prior to use if out of service for greater than one year. The licensee utilized PM1008323 to perform the UMAX VVM in-service inspection, UMAX VVM external surface visual examination, UMAX VVM inspection of visual markings, and to perform the ISFSI pad inspection. The licensee utilized PM1008325 to perform the HI-TRAC VW pressure relief valve calibration.

Documents Reviewed: a) Callaway Procedure PM1008323, "UMAX and ISFSI pad maintenance," Revision 0; b) Callaway Procedure PM1008325, "HI-TRAC VW pressure relief valve calibration," Revision 0

Category: Loading Operations **Topic:** Cask System Inspections Prior to Use

Reference: FSAR 1040, Table 10.4.1 Revision 2

Requirement The following HI-STORM UMAX areas shall be inspected prior to MPC installation: a) CEC cavity visual inspection; b) divider shell visual inspection; c) closure lid examination.

Observation: The licensee had incorporated the UMAX FSAR requirements into their procedures to perform the required inspections prior to MPC loading for; the CEC cavity visual inspection, divider shell visual inspection, and closure lid examination. The licensee utilized Procedure ETP-ZZ-04021 for all three inspections. Procedure ETP-ZZ-04021 contained the inspection requirements listed in FSAR 1040 Table 10.4.1, HI-Storm System Maintenance Program Schedule, for all three components.

Documents Reviewed: a) Callaway Procedure ETP-ZZ-04021, "ISFSI Loading Campaign Performance - IPTE," Revision 1

Category: Loading Operations **Topic:** Fuel Cladding Not Exposed to Air

Reference: CoC 1040, Appendix B, Section 3.4.13 Amendment 0

Requirement Procedures and/or mechanical barriers shall be established to ensure that during loading operations (and unloading) that either the fuel cladding is covered by water or the canister is filled with an inert gas.

Observation: This criteria was met at Callaway. Use of special equipment and procedures ensured the fuel was not exposed to air during loading operations. Procedures for loading and

unloading dry fuel storage canisters at Callaway all included a caution statement to ensure that the water level in the canister was not lowered below the top of the fuel cladding to avoid exposing fuel to the atmosphere, to prevent oxidation and potential fuel damage. In this regard, Holtec/Callaway had established procedures which ensured that during loading operations that either the fuel cladding was covered by water or the canister was filled with an inert gas. Special equipment such as the "dip tube" and flow meters ensured that less than 50 gallons of water would be removed to support welding operations while also preventing the fuel from being exposed to air.

Documents Reviewed: a) Holtec Procedures HPP-2253-200, "MPC Loading at Callaway," Revision 9; b) Holtec Procedure HPP-2253-500, "MPC Unloading at Callaway," Revision 4

Category: Loading Operations **Topic:** Handling Damaged Fuel Containers

Reference: FSAR 1032, Sections 2.1.3 and 6.4.4.1 Revision 3

Requirement Damaged fuel assemblies and fuel debris shall be loaded into damaged fuel containers (DFCs) prior to being loaded into the canister.

Observation: Damaged Fuel Containers were not to be used during the initial loading campaigns. Section 3.0 of the Callaway Procedure ETP-ZZ04020 provided the Acceptance and Functional Criteria for loading operations. The procedure stated that only fuel assemblies meeting the requirements of the HI-STORM, UMAX Certificate of Compliance for undamaged assemblies were to be loaded into spent fuel canisters.

Documents Reviewed: a) Callaway Procedure ETP-ZZ04020, "Fuel Selection and Cask Loading for Dry Cask Storage," Revision 0

Category: Loading Operations **Topic:** MPC-37 Boron Concentration

Reference: CoC 1014, Tech Spec A.3.3.1 and associated tables Amendment 0

Requirement The boron concentration in the canister shall meet the limits specified in Technical Specification 3.3.1 for the applicable canister model and the most limiting fuel assembly array and class. The boron concentration must be verified within required limits using two independent measurements taken within four hours of filling the canister with water, and every 48 hours thereafter while fuel and water are in the canister.

Observation: The boron concentration sampling requirements specified in the Technical Specification were placed in the implementing procedures for ISFSI loading operations at Callaway. Callaway's loading and unloading procedures HPP-2253-200 and HPP-2253-500 both contained the appropriate steps to sample for boron concentrations when loading fuel assemblies, adding water from the pool during hydrostatic tests, and prior to MPC alternate cooling operations.

Documents Reviewed: a) Holtec Procedure HPP-2253-500 "MPC Unloading at Callaway," Revision 7; b) Holtec Procedure HPP-2253-200 "MPC Loading at Callaway," Revision 9

Category: Loading Operations **Topic:** Pressure Relief Valves

Reference: FSAR 1032, Table 9.1.1 Revision 3

Requirement Pressure relief valves in the water and gas processing systems limit the canister pressure to acceptable levels.

Observation: Pressure relief valves were required and utilized during water and gas processing system operations at the Callaway site. A review of Holtec's procedure for MPC sealing at Callaway showed two pressure relief valves in the schematic diagram present in Attachment 8.10. NRC inspectors verified the presence of these pressure relief values during the second dry-run, June 2-4, 2015, and during initial fuel loading operations at Callaway. The safety relief valves were set at two different pressure values, 95 and 140 psig. The design pressure of the MPC was 100 psig. The 95 psig pressure relief valve was utilized throughout the canister drying and backfilling process. The 140 psig pressure relief valve was utilized during the hydrostatic testing of the MPC when the pressure of the system was raised to 125 psig.

Documents Reviewed: a) Holtec Procedure HPP-2253-300, "MPC Sealing at Callaway," Revision 7

Category: Loading Operations **Topic:** Time-to-Boil Time Limits

Reference: FSAR 1032, Section 4.5.3 and Table 4.5.4 Revision 3

Requirement Wet transfer operations begin when the lid is placed on the canister in the spent fuel pool and end when the canister is blown down following pressure testing. During wet operations, the water inside the MPC is not permitted to boil. Using the design basis heat load, Table 4.5.4 of the FSAR provides the time-to-boil for various initial water temperatures. If wet transfer operations cannot be completed prior to boiling, a forced water circulation shall be initiated and maintained to remove decay heat from the canister cavity. The minimum water flow rate required to maintain the MPC cavity water temperature below boiling with an adequate subcooling margin is determined using a calculation provided in Section 4.5.3 of the FSAR.

Observation: Callaway had procedure steps in place that adequately calculated and controlled the time to boil limitations put into place by the Holtec FSAR. Holtec's MPC loading procedure for Callaway included a time to boil calculation that was based on the HI-STORM FW FSAR. The HI-STORM FW used the same MPC design as the Holtec UMAX at Callaway. The calculation for the time to boil appeared in Step 7.6.40 of the loading procedure, HPP-2253-200. In addition, the MPC sealing procedure included steps that precluded draining the HI-TRAC/MPC annulus until the MPC blow down was to begin. Adherence to the time to boil clock, assures that the fuel assemblies and cladding would be maintained at acceptable temperatures until just prior to the start of force helium dehydration, which also controlled cladding temperatures until the helium backfill has been completed.

If the time to boil limit could not be met, Callaway's procedure HPP-2253-300 Attachment 8.13, "Contingency Steps for MPC Alternate Cooling," required establishing a forced water circulation with minimum flow rate of 11 gallons per minute that was calculated in accordance with FSAR Section 4.5.3.

Documents Reviewed: a) Holtec Procedure HPP-2253-200, "MPC Loading at Callaway," Revision 9; b) Holtec Procedure HPP-2253-300, "MPC Sealing at Callaway," Revision 7

Category: Loading Operations **Topic:** Transfer Cask Inspections Prior to Use

Reference: FSAR 1040, Table 10.4.1 Revision 2

Requirement The following HI-TRACK VM transfer cask area shall be inspected prior to use: a) HI-TRAC cavity visual inspection; b) HI-TRAC TAL visual inspection; c) HI-TRAC bottom lid bolts and bolt holes; d) HI-TRAC water jacket visual verification.

Observation: The licensee had incorporated the UMAX FSAR requirements into Procedure ETP-ZZ-04021 and Preventive Maintenance (PM) job order PM 1008270. The documents performed the required HI-TRAC VW cavity visual inspection, HI-TRAC Threaded Anchor Location (TAL) visual inspection, HI-TRAC bottom lid bolts & bolt holes inspection, and HI-TRAC waterjacket visual verification prior to HI-TRAC VW use.

Documents Reviewed: a) Callaway Procedure ETP-ZZ-04021, "ISFSI Loading Campaign Performance - IPTE," Revision 1; b) Callaway Procedure PM1008270, "Perform HI-Trac Inspection per HI-STORM FW FSAR Table 9.2.5 and UMAX FSAR Section 10.4.1," Revision 0

Category: NDE-Helium Leak Testing **Topic:** Helium Leak Test-Vent/Drain Covers

Reference: CoC 1040, Tech Spec A.3.1.1.3 Amendment 0

Requirement The helium leak rate through the canister vent and drain port confinement welds shall meet the leak tight criteria of ANSI N14.5 (1997). This degree of containment is achieved by demonstration of a leakage rate less than or equal to 2 X 10(-7) atm·cc/sec of helium at an upstream pressure of 1 atmosphere (atm) absolute (abs) and a downstream pressure of 0.01 atm abs or less.

Observation: The helium leak test was demonstrated during the welding dry run May 19-21, 2015 consistent with the acceptance standards specified in Certificate of Compliance 1040 and ANSI N14.5-1997. Procedure MSLT-MPC-Holtec, Step 8.1 required that the total leak rate of the vent port plus the drain port be less than or equal to 2 x 10(-7) atm cc/sec helium. Additionally, NRC inspectors witnessed the helium leak rate test on the first canister loaded on September 1, 2015. The leak rate was verified to be below the Technical Specification limit.

Documents Reviewed: a) MSLT-MPC-Holtec, "Helium Mass Spectrometer Leak Test Procedure MPC," Revision Callaway-01

Category: NDE-Helium Leak Testing **Topic:** HMSLD Minimum Sensitivity

Reference: ANSI N14.5, Section 8.4 Published 1997

Requirement The helium mass spectrometer leak detector (HMSLD) shall have a minimum sensitivity of 1/2 the acceptance leak rate. For example, a package with a leak tight acceptance criteria of 1.0 X 10(-7) ref-cc/sec requires a minimum helium mass spectrometer leak detector sensitivity of 5.0 x 10(-8) ref-cc/sec. This sensitivity requirement applies to both the hood and detector probe methods. The helium mass spectrometer leak detector shall be calibrated to a traceable standard.

Observation: The helium mass spectrometer leak detector (MSLD) minimum sensitivity requirement was specified in Callaway's procedure. Procedure MSLT-MPC-Holtec, Step 4.1 required the MSLD sensitivity to be less than 1/2 the acceptance criteria leak rate. The minimum acceptable leak rate was stated in Section 8.2 as 2.0 x 10(-7) atm-cc/sec (He) and referenced Technical Specification 3.1.1.3. Technical Specification 3.1.1.3 required the helium leak rate through the canister vent and drain port confinement welds to meet the leak tight criteria specified in ANSI N14.5-1997. Section 2, "Definitions," of ANSI N14.5-1997 defined leak tight as having a leak rate of 1 x 10(-7) ref-cc/sec of air. A note to the definition stated that 1 x 10(-7) ref-cc/sec of air was equivalent to 2 x 10(-7) atm-cc/sec of helium. A calibration standard traceable to the National Institute of Standards and Technology (NIST) with a leak rate between 10(-6) and 10(-8) atm-cc/sec was required by Step 4.3 of Procedure MSLT-MPC-Holtec. Section 5.0, "MSLD Startup and Instrument Calibration," provided instructions on calibrating the helium mass spectrometer. On September 1, 2015, NRC inspectors observed the qualified non-destructive examiner properly set up the MSLD per Callaway's approved procedure and perform the helium leak test.

Documents Reviewed: a) MSLT-MPC-Holtec, "Helium Mass Spectrometer Leak Test Procedure MPC," Revision Callaway-01

Category: NDE-Liquid Penetrant **Topic:** Acceptance Criteria

Reference: ASME Section III, Article NB-5352 Published 2007

Requirement Only indications with major dimensions greater than 1/16 inch should be considered relevant. The following relevant indications are unacceptable: (1) any cracks or linear indications. Linear indications have a length at least 3 times greater than the width; (2) rounded indications with dimensions greater than 3/16 inch (5 mm); (3) more than four rounded indications in a line, separated by 1/16 inch (1.5 mm) or less edge to edge; and (4) ten or more rounded indications in any 6 square inch area in the most unfavorable location relative to the indications being evaluated.

Observation: The acceptance standards from ASME Section III are found in Appendix A, "Acceptance Standards," of the PCI liquid dye penetrant procedure, GQP-9.2. These criteria were fully met during the welding dry-run at Callaway on May 19-21, 2015 and during the first fuel loading at Callaway, which began on August 24, 2015.

Documents Reviewed: a) PCI General Quality Procedure, GQP-9.2, "High Temperature Liquid Penetrant Examination and Acceptance Standards for Welds, Base Materials, and Cladding," Revision 8; b) PCI General Quality Procedure, GQP-9.7, "Liquid Penetrant Examination and Acceptance Standards for Welds, Base Materials, and Cladding," Revision 11

Category: NDE-Liquid Penetrant **Topic:** Contaminants

Reference: ASME Section V, Article 6, T-641 Published 2007

Requirement The user shall obtain certification of contaminant content for all liquid penetrant materials used on austenitic stainless steels. The certifications shall include the manufacturers batch number and sample results. Sub-article T-642 limits the total halogen (chlorine plus fluorine) content of each agent (penetrant, cleaner and developer) to 1.0 weight percent (wt.%) when used on austenitic stainless steels.

Observation: NRC inspectors reviewed vendor supplied laboratory test results for the liquid dye penetrant, developer, and cleaner used for non-destructive testing at Callaway and verified that the total halogen content of each agent did not exceed the ASME Section V, Article 6, T-641 requirement of 1% (by weight) for use on austenitic stainless steels.

Documents Reviewed: a) Sherwin Incorporated Certification for DUBL-CHEK KO-19, Batch No. 06-B56; b) Sherwin Incorporated Certification for DUBL-CHEK KO-17, Batch No. 07-B54; c) Sherwin Incorporated Certification for DUBL-CHEK D-350, Batch No. 99-L71

Category: NDE-Liquid Penetrant **Topic:** Final Interpretation

Reference: ASME Section V, Article 6, T-676.1 Published 2007

Requirement Final interpretation shall be made after allowing the penetrant to bleed-out for 10-60 minutes under standard temperatures (50 and 125 degrees F). The 10-60 minute clock starts immediately after application of a dry developer.

Observation: PCI's liquid dye penetrant procedure, GQP-9.7, had the requirements of ASME Section V, Article 6, T-676.1 spelled out in step 9.7.1(a). For high temperature uses, procedure GQP-9.2 will be used. GQP-9.2 was used during welding dry-run activities at the Callaway site. The manufacturer's instructions (Sherwin Incorporated) for the high temp penetrant, KO-17, high temperature cleaner, KO-19, and high temperature developer, D-350, listed different developer dwell times based on the surface temperature of the area being examined. The use of the Sherwin high temperature liquid penetrant products was fully qualified by PCI for use during dry fuel storage operations at Callaway. The manufacturer's recommendations supersede general code requirements in situations such as this one.

Documents Reviewed: a) PCI General Quality Procedure GQP-9.7, "Liquid Penetrant Examination and Acceptance Standards for Welds, Base Materials, and Cladding," Revision 11; b) PCI GQP-9.2, "High Temperature Liquid Penetrant Examination and Acceptance Standards for Welds, Base Materials, and Cladding," Revision 8

Category: NDE-Liquid Penetrant **Topic:** Lid-To-Shell Weld PT

Reference: CoC 1040, Appendix B, Table 3-1 Amendment 0

Requirement Only ultrasonic testing or multi-layer liquid penetrant (PT) examination is permitted on the lid-to-shell weld. If PT alone is used, at a minimum, it will include the root and final weld layers and each approximately 3/8 inch of weld depth.

Observation: PCI's Project Instruction called out the requirement that liquid penetrant (PT) examination shall be performed after the root layer and every layer of weld material afterwards. The layer was defined as equal to or less than 3/8 inch thickness (consisting of one or more passes), as referenced in the Holtec CoC 1040, Appendix B, Table 3-1. This requirement was met in the procedure and observed in practice during the welding dry-run activities at Callaway the week of May 18, 2015.

Documents Reviewed: a) PCI Project Instruction PI-CNSTR-OP-H-01, "Closure Welding of Holtec Multi-Purpose Canisters - UMAX," Revision 0

Category: NDE-Liquid Penetrant **Topic:** Minimum Elements

Reference: ASME Section V, Article 6, T-621 Published 2007

Requirement Each liquid penetrant (PT) procedure shall include the requirements listed in Table T-621: (1) type of each penetrant, remover, emulsifier, and developer; (2) Surface preparation (finishing and cleaning, including type of cleaning solvent); (3) Method of applying penetrants; (4) Method of removing excess surface penetrant; (5) Hydrophilic or lipophilic emulsifier concentration and dwell time in dip tanks and agitation time for hydrophilic emulsifiers; (6) Hydrophilic emulsifier concentration in spray applications; (7) Method of applying developer; (8) Minimum and maximum time periods between steps and drying aids; (9) Decrease in penetrant dwell time; (10) Increase in developer dwell time (Interpretation Time); (11) Minimum light intensity; (12) Surface temperature outside 40 degrees F to 125 degrees F or as previously qualified; (13) Performance demonstration, when required; (14) Personnel qualification requirements; (15) Materials, shapes, or sizes to be examined and the extent of examination; and (16) Post-examination cleaning technique.

Observation: The two procedures in use by PCI both contain all of the applicable minimum elements as specified in ASME Section V with the following allowed exceptions: Elements 5 and 6, both dealing with hydrophilic or lipophilic emulsifier concentration, did not apply to the specific types of dye penetrants and developers used for NDE at Callaway. Also, the temperature range of element #12 was not met, 40 to 125 degrees F. Instead, the solvent, dye, and developer used at Callaway by PCI were qualified for use from 50 to 350 degrees F. PCI and Callaway met the code required minimum elements in their liquid dye penetrant procedures.

Documents Reviewed: a) PCI General Quality Procedure GQP-9.2, "High Temperature Liquid Penetrant Examination and Acceptance Standards for Welds, Base Materials, and Cladding," Revision 8; b) PCI General Quality Procedure GQP-9.7, "Liquid Penetrant Examination and Acceptance Standards for Welds, Base Materials, and Cladding," Revision 11

Category: NDE-Liquid Penetrant **Topic:** Removing Excess Penetrant

Reference: ASME Section V, Article 6, T-673.3 Published 2007

Requirement Excess solvent removable penetrants shall be removed by wiping with a cloth or absorbent paper until most traces of the penetrant have been removed. The remaining traces shall be removed by lightly wiping the surface with a cloth or absorbent paper moistened with solvent. Care shall be taken to avoid the use of excess solvent. Flushing the surface with solvent, following the application of the penetrant and prior to developing, is prohibited.

Observation: The PCI Liquid Dye Penetrant procedure was compliant with the requirements of ASME Section V, Article 6, T-673.3, but included provisions for spraying the cleaner solvent directly onto the surface of the area to be inspected. Consultation with the manufacturer's recommendations for Dubl-Chek KO-19 cleaner indicated that this was an acceptable practice. Further consultation with HQ and an industry Level III NDE technician indicated that since KO-19 was applied as a foam, spraying it directly onto the surface to be examined does not constitute "flushing the surface" with solvent. PCI was fully compliant with the ASME requirements and manufacturers recommendations for use and removal of excess dye penetrant.

Documents Reviewed: a) PCI General Quality Procedure, GQP-9.2, "High Temperature Liquid Penetrant Examination and Acceptance Standards for Welds, Base Materials, and Cladding," Revision 8; b) Sherwin Incorporated KO-19 Hi-Temp Remover Technical Data Sheet

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Category: NDE-Liquid Penetrant **Topic:** Surface Preparation

Reference: ASME Section V, Article 6, T-642 (b) Published 2007

Requirement Prior to each liquid penetrant examination, the surface to be examined and all adjacent areas within one inch must be dry and free of all dirt, grease, lint, scale, welding flux, weld spatter, paint, oil, and other extraneous matter that could obscure surface openings or otherwise interfere with the examination.

Observation: PCI's liquid penetrant procedure included the requirement that the surface to be examined be free of all dirt, grease, link, scale, welding flux, weld spatter, and other contaminants on the surface in step 9.1.1.b. PCI's procedure was fully compliant with the requirements listed in ASME Section V, Article 6, T-642(b).

Documents Reviewed: a) PCI General Quality Procedure, GQP-9.2, "High Temperature Liquid Penetrant Examination and Acceptance Standards for Welds, Base Materials, and Cladding," Revision 8

Category: NDE-Personnel Qualification **Topic:** Certification Records

Reference: SNT-TC-1A, Section 9 Published 1992

Requirement Certification records should contain the name of the certified individual, the certification level and method, the individual's educational background and NDE experience, a statement of satisfactory completion of training per the employer's written practice, visual examination results, evidence of successful completion of examinations including grades, date of certification, and the signature of the employer.

Observation: NRC inspectors reviewed the training, testing, and certification records for two Level II NDE technicians from PCI. One of the technicians was involved in the dry-run activities witnessed during the week of May 18, 2015. The other individual was the one who actually participated at Callaway during their initial dry fuel storage campaign. The records for both individuals were complete and included all of the required information as specified by SNT-TC-1A, Section 9.

Documents Reviewed: a) PCI Visual Examination Report; b) PCI NDE VT Level II Personnel Certificate; c) PCI NDE PT Level II Personnel Certificate; and d) PCI Certification of Inspection, Examination, and Testing Personnel

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Category: NDE-Personnel Qualification **Topic:** Level II Exam Grading

Reference: SNT-TC-1A, Section 8 Published 1992

Requirement Level II technicians take 3 examinations: Basic, Method, and Specific. A composite grade should be determined by simple averaging of the results of the 3 examinations. A passing composite grade should be 80% with no examination results below 70%.

Observation: NRC inspectors reviewed the PCI procedure that directs the training, qualification, examination, and certification of NDE personnel. The requirements within the

procedure contained the requirements of SNT-TC-1A, Section 8. NRC inspectors verified that the Level II technician had passed all of his composite and individual examination requirements with an acceptable level of accuracy and proficiency. This requirement was met.

Documents Reviewed: a) PCI GQP-9.0 "Training, Qualification, Examination, and Certification of NDE Personnel in Accordance with SNT-TC-1A and CP-189," Revision 15; b) various NDE VT Level II Personnel Certificates, NDE PT Level II Personnel Certificates, and a Certificate of Inspection, Examination, and Testing Personnel for the individual who performed VT and PT for the welding dry-run at Callaway

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Category: NDE-Personnel Qualification **Topic:** Level III Candidates

Reference: SNT-TC-1A, Section 6 Published 1992

Requirement A Level III candidate who has completed less than 2 years of engineering or science study must have 4 years of experience comparable to a Level II. A Level III candidate who has completed 2 years of engineering or science study must have 2 years of experience comparable to a Level II. A Level III candidate who has completed 4 years of engineering or science study must have 1 year of experience comparable to a Level II.

Observation: The welding contractor for Callaway, PCI, had these requirements clearly spelled out in its procedure for training and qualification of NDE personnel. Step 7.5.2 of the procedure laid out the education, training, and relevant experience requirements for Level III technicians. PCI had implemented Level III requirements that were fully compliant with SNT-TC-1A, Section 6.

Documents Reviewed: a) PCI Procedure GQP-9.0, "Training, Qualification, Examination, and Certification of NDE Personnel in Accordance with SNT-TC-1A and CP-189," Revision 15

Category: NDE-Personnel Qualification **Topic:** Recertification of Personnel

Reference: SNT-TC-1A, Section 9 Published 1992

Requirement Maximum recertification intervals are 3 years for Levels I and II, and 5 years for Level III. Recertification may be granted without testing provided there is documented continuing satisfactory performance. Without documented continuing satisfactory performance, reexamination is required for those sections deemed necessary by the Level III examiner.

Observation: PCI's procedure, GQP-9.0, for training and qualification of NDE personnel covered recertification requirements in Steps 11.2.1 through 11.2.7. The requirements of SNT-TC-1A, Section 9 were met by PCI's written procedure. PCI's procedure does not address Level I personnel. However, no Level I personnel were supporting dry fuel storage activities at Callaway.

Documents Reviewed: a) PCI Procedure, GQP-9.0, "Training, Qualification, Examination, and Certification of NDE Personnel in Accordance with SNT-TC-1A and CP-189," Rev. 15

Category: NDE-Personnel Qualification **Topic:** Visual Acuity

Reference: SNT-TC-1A, Section 8.2 Published 1992

Requirement The NDE examiner should have natural or corrected near-distance acuity in at least one eye capable of reading Jaeger Number 1 at a distance of not less than 12 inches on a standard Jaeger test chart, or capable of perceiving a minimum of 8 on an Ortho-Rater test pattern. This should be verified annually. The NDE examiner should demonstrate the capability of distinguishing and differentiating contrast among colors used in the applicable method. This should be verified every 3 years.

Observation: PCI Procedure GQP-9.0 required visual acuity examinations for NDE personnel in accordance with SNT-TC-1A. The overarching procedure, GQP-9.0 established the yearly frequency for the visual examination of the NDE personnel. GQP-9.14 established a procedure for carrying out the visual examination which included near and far distance visual acuity and color discrimination. PCI had adequate instructions and procedures in place to support qualification of personnel to perform NDE examinations. NRC also inspected the visual evaluation reports for PCI staff NDE personnel. Dry fuel operations at Callaway met this requirement.

Documents Reviewed: a) PCI General Quality Procedures GQP-9.14, "Visual Acuity Examinations," Revision 2; b) PCI GQP-9.0, "Training, Qualification, Examination, and Certification of NDE Personnel in Accordance with SNT-TC-1A and CP-189," Revision 15; c) PCI Visual Evaluation Report, SAP# 73566

Category: NDE-Personnel Qualification **Topic:** Written Practice

Reference: SNT-TC-1A, Section 5 Published 1992

Requirement The employer shall establish a written practice for control and administration of non-destructive testing personnel training, examination, and certification. The written practice should describe the responsibility of each level of certification for determining the acceptability of material or components. The written practice shall describe the training experience and examination requirements for each level of certification.

Observation: PCI Procedure GQP-9.0 established the written requirements for NDE personnel qualifications and included sections that address personnel certification levels, responsibilities, education, experience requirements, training, visual acuity examinations, and certification examinations. PCI handing of personnel qualifications was fully compliant with SNT-TC-1A, Section 6.

Documents Reviewed: a) PCI Procedure GQP-9.0, "Training, Qualification, Examination, and Certification of NDE Personnel in Accordance with SNT-TC-1A and CP-189," Revision 15

Category: NDE-Visual Examination **Topic:** Acceptance Criteria - Arc Strikes

Reference: ASME Section III, Article NF-5360 (i) Published 2007

Requirement Arc strikes and blemishes in the weld or base material are acceptable, provided no cracking is visually detected.

Observation: The arc strike acceptance criteria was incorporated into the visual weld examination procedure. Procedure GQP-9.6 Addendum 1 included the requirements of the ASME

Section III, Article NF-5360, which was utilized by the NDE inspector during visual examinations to verify the acceptance criteria was met.

Documents Reviewed: a) PCI Procedure GQP-9.6 "Visual Examination of Welds," Revision 14

Category: NDE-Visual Examination **Topic:** Acceptance Criteria - Cracks

Reference: ASME Section III, Article NF-5360 (a) Published 2007

Requirement Cracks are unacceptable.

Observation: The crack acceptance criteria was incorporated into the visual weld examination procedure. Procedure GQP-9.6 Addendum 1 included the requirements of the ASME Section III, Article NF-5360, which was utilized by the NDE inspector during visual examinations to verify the acceptance criteria was met.

Documents Reviewed: a) PCI Procedure GQP-9.6, "Visual Examination of Welds," Revision 14

Category: NDE-Visual Examination **Topic:** Acceptance Criteria - Craters

Reference: ASME Section III, Article NF-5360 (e) Published 2007

Requirement Craters outside the weld area are irrelevant, provided there are no cracks.

Observation: The craters acceptance criteria was incorporated into the visual weld examination procedure. Procedure GQP-9.6, Addendum 1 included the requirements of the ASME Section III, Article NF-5360, which was utilized by the NDE inspector during visual examinations to verify the acceptance criteria was met.

Documents Reviewed: a) PCI Procedure GQP-9.6, "Visual Examination of Welds," Revision 14

Category: NDE-Visual Examination **Topic:** Acceptance Criteria - Fusion

Reference: ASME Section III, Article NF-5360 (c) Published 2007

Requirement For fillet welds, incomplete fusion of more than 3/8" (10 mm) in any 4" (100 mm) segment is unacceptable. For fillet welds, incomplete fusion of more than 1/4" (6 mm) in welds less than 4" (100 mm) is unacceptable. For groove welds, any incomplete fusion is unacceptable. Rounded end conditions (starts and stops) shall not be considered indications of incomplete fusion.

Observation: The fusion acceptance criteria was incorporated into the visual weld examination procedure. Procedure GQP-9.6 Addendum 1 included the requirements of the ASME Section III, Article NF-5360, which was utilized by the NDE inspector during visual examinations to verify the acceptance criteria was met.

Documents Reviewed: a) PCI Procedure GQP-9.6, "Visual Examination of Welds," Revision 14

Category: NDE-Visual Examination **Topic:** Acceptance Criteria - Lengths

Reference: ASME Section III, Article NF-5360 (h) Published 2007

Requirement For welds 3" and longer, weld lengths shorter than specified by more than 1/4" (6 mm) are unacceptable. For welds less than 3" long, weld lengths shorter than specified by

more than 1/8" (3.2 mm) are unacceptable. Intermittent welds not spaced within 1" (25 mm) of the specified location are unacceptable.

Observation: The weld length acceptance criteria was incorporated into the visual weld examination procedure. Procedure GQP-9.6 Addendum 1 included the requirements of the ASME Section III, Article NF-5360, which was utilized by the NDE inspector during visual examinations to verify the acceptance criteria was met.

Documents Reviewed: a) PCI Procedure GQP-9.6, "Visual Examination of Welds," Revision 14

Category: NDE-Visual Examination **Topic:** Acceptance Criteria - Overlap

Reference: ASME Section III, Article NF-5360 (d) Published 2007

Requirement When fusion in the overlap length cannot be verified, an overlap length of greater than 3/8" (10 mm) in any 4" (100 mm) segment, and 1/4" (6 mm) in welds less than 4" (100 mm) long, is unacceptable.

Observation: The overlap acceptance criteria was incorporated into the visual weld examination procedure. Procedure GQP-9.6 Addendum 1 included the requirements of the ASME Section III, Article NF-5360, which was utilized by the NDE inspector during visual examinations to verify the acceptance criteria was met.

Documents Reviewed: a) PCI Procedure GQP-9.6, "Visual Examination of Welds," Revision 14

Category: NDE-Visual Examination **Topic:** Acceptance Criteria - Porosity

Reference: ASME Section III, Article NF-5360 (g) Published 2007

Requirement The following degrees of random porosity are unacceptable: (1) the sum of the diameters of random porosity exceeding 3/8" (10 mm) in any one linear inch of weld; (2) the sum of the diameters of random porosity exceeding 3/4" (19 mm) in any 12 linear inches (305 mm) of weld; or (3) four or more pores aligned, and the pores separated by 1/16" (1.6 mm) or less edge to edge.

Observation: The porosity acceptance criteria was incorporated into the visual weld examination procedure. Procedure GQP-9.6 Addendum 1 included the requirements of the ASME Section III, Article NF-5360, which was utilized by the NDE inspector during visual examinations to verify the acceptance criteria was met.

Documents Reviewed: a) PCI Procedure GQP-9.6, "Visual Examination of Welds," Revision 14

Category: NDE-Visual Examination **Topic:** Acceptance Criteria - Slag

Reference: ASME Section III, Article NF-5360 (j) Published 2007

Requirement Slag 1/8" (3.2 mm) or less in size is irrelevant. Slag greater than 1/4" (6 mm) in size after cleaning is unacceptable.

Observation: The slag acceptance criteria was incorporated into the visual weld examination procedure. Procedure GQP-9.6 Addendum 1 included the requirements of the ASME Section III, Article NF-5360, which was utilized by the NDE inspector during visual examinations to verify the acceptance criteria was met.

Documents Reviewed: a) PCI Procedure GQP-9.6, "Visual Examination of Welds," Revision 14

Category: NDE-Visual Examination **Topic:** Acceptance Criteria - Thickness

Reference: ASME Section III, Article NF-5360 (b) Published 2007

Requirement Welds thinner than specified by greater than 1/16" (1.6 mm) for more than one-fourth the weld length are unacceptable. Welds thicker than specified are unacceptable if they interfere with mating parts.

Observation: The weld thickness acceptance criteria was incorporated into the visual weld examination procedure. Procedure GQP-9.6 Addendum 1 included the requirements of the ASME Section III, Article NF-5360, which was utilized by the NDE inspector during visual examinations to verify the acceptance criteria was met.

Documents Reviewed: a) PCI Procedure GQP-9.6, "Visual Examination of Welds," Revision 14

Category: NDE-Visual Examination **Topic:** Acceptance Criteria - Undercut

Reference: ASME Section III, Article NF-5360(f)(2) Published 2007

Requirement Undercuts deeper than 1/32" (.8 mm) on one side for the full length of the weld are unacceptable. Undercuts deeper than 1/32" (.8 mm) on one side for one-half the length of the weld AND deeper than 1/16" (1.6 mm) on the same side for one-fourth the length of the weld, are unacceptable.

Observation: The undercut acceptance criteria was incorporated into the visual weld examination procedure. Procedure GQP-9.6 Addendum 1 included the requirements of the ASME Section III, Article NF-5360, which was utilized by the NDE inspector during visual examinations to verify the acceptance criteria was met.

Documents Reviewed: a) PCI Procedure GQP-9.6, "Visual Examination of Welds," Revision 14

Category: NDE-Visual Examination **Topic:** Eye Position and Lighting

Reference: ASME Section V, Article 9, T-952 Published 2007

Requirement Direct visual examinations shall be conducted with the eye within 24" (610 mm) of the surface, at an angle not less than 30 degrees. The minimum light level shall be 100 foot-candles.

Observation: The PCI procedure used at Callaway, Procedure GQP-9.6, was verified by NRC inspectors to contained the ASME Section V, Article 9, T-952 requirement. PCI Procedure GQP-9.6, Step 4.1 stated "Direct visual examinations are those which can be made when access is sufficient to place the eye within 24 inches of the surface and at an angle not less than 30 degrees to the surface to be examined. Mirrors may be used to improve the angle of vision, and aids such as a magnifying lens may be used to assist examinations." Step 6.2 of the procedure stated "For direct visual examination, lighting shall be provided such that the specific part, component, vessel or section thereof, under immediate examination is illuminated to attain a minimum of 100 foot-candles. Illumination may be by any means including a hand held flashlight."

Documents Reviewed: a) PCI Procedure GQP-9.6, "Visual Examination of Welds," Revision 14

Category: NDE-Visual Examination **Topic:** Minimum Elements

Reference: ASME Section V, Article 9, T-921.1 Published 2007

Requirement Each Visual Testing (VT) procedure shall include the: (1) Technique used either direct or remote; (2) Remote visual aids; (3) Personnel performance requirements, when required; (4) Lighting intensity; (5) Configurations to be examined and base material product forms(pipe, plate, forgings, etc.); (6) Lighting equipment; (7) Methods or tools used for surface preparation; (8) Equipment or devices used for a direct technique; (9) sequence of examination; (10) Personnel qualifications.

Observation: The PCI procedure used at Callaway, GQP-9.6, was verified by NRC inspectors to contained the required elements specified by ASME Section V, Article 9, T-921.1 related to visual examination (testing) requirements.

Documents Reviewed: a) PCI Procedure GQP-9.6, "Visual Examination of Welds," Revision 14

Category: NDE-Visual Examination **Topic:** Procedure Validation

Reference: ASME Section V, Article 9, T-921.3 Published 2007

Requirement The procedure shall contain or reference a report of what was used to demonstrate that the examination procedure was adequate. In general, a fine line 1/32" or less in width, or some other artificial flaw located on the surface or a similar surface to that to be examined, may be considered a test method for this demonstration. The line or artificial flaw should be in the least discernible location on the area examined, to prove the procedure.

Observation: NRC inspectors reviewed procedure qualification record for the PCI Procedure GQP-9.6 used at Callaway, that demonstrated the visual examination procedure was adequate. PCI Letter "Visual Examination Demonstration to HSB Global Standards ANI of GQP 9.6 Revision 14," dated December 4, 2013, documented the qualification of PCI's procedure. Attached to the letter was the visual testing examination report form that documented the qualification results.

Documents Reviewed: a) PCI Procedure GQP-9.6, "Visual Examination of Welds," Revision 14; b) PCI Letter, "Visual Examination Demonstration to HSB Global Standards ANI of GQP 9.6 Revision 14," dated December 4, 2013

Category: Pressure Testing **Topic:** Governing Code

Reference: FSAR 1032, Section 10.1.2.2.2 Revision 3

Requirement Pressure testing (hydrostatic or pneumatic) of the canister confinement boundary shall be performed in accordance with the requirements of ASME Code Section III, Subsection NB, Article NB-6000, when field welding of the canister lid-to-shell weld is completed. If hydrostatic testing is used, the canister shall be pressure tested to 125% of design pressure.

Observation: The requirements for the canister hydrostatic testing had been incorporated into Procedure HPP-2253-300 consistent with ASME Code Section III, Subsection NB,

Article NB-6000. The canister design pressure was 100 pounds per square inch-gauge (psig) per Holtec UMAX FSAR, Table 2.3.5 "Design (Maximum Allowable) Pressures." Procedure HPP-2253-300, Section 8.4, "Hydrostatic Test with FHD" provided the instructions for performing the canister hydrostatic test. Procedure HPP-2253-300, Step 7.4.30 required pressurizing the canister to 125.5-129.5 psig. Test results were documented in Step 7.4.41 after the pressure was maintained for 10 minutes. Procedure Step 7.4.42 required an Ameren Representative to witness the hydrostatic test and sign off on the test results.

Documents Reviewed: a) Holtec Procedure, HPP-2253-300 "MPC Sealing at Callaway," Revision 7

Category: Pressure Testing **Topic:** Hydrostatic Testing Sequence

Reference: FSAR 1032, Table 2.2.1, Sections 9.2.5;10.1.2.2.2 Revision 3

Requirement During hydrostatic testing, demineralized water or spent fuel pool water is admitted to the canister through a supply line connected to the drain port RVOA. The canister is pressurized to 125 +5/-0 psig and held for 10 minutes with no pressure drop. Following the 10-minute hold at test pressure, the canister lid to shell weld is examined to confirm no observable water leakage. The canister is then depressurized through a return line connected to the vent port RVOA and routed back to the spent fuel pool or liquid radwaste system. Once the canister is depressurized, the liquid penetrant examination of the canister lid-to-shell weld is repeated. Any evidence of cracking or deformation is cause for rejection.

Observation: NRC inspectors verified that the hydrostatic testing was performed properly with a positive displacement pump. The technician maintained the pressure in the correct pressure range (125.5 – 129.5 psi) for the ten minutes prescribed by the testing procedure. NRC observed the hydrostatic test during dry-run #2 June 2-4, 2015.

Documents Reviewed: a) Holtec Procedures HPP-2253-300, "MPC Sealing at Callaway," Revision 7

Category: Pressure Testing **Topic:** Pressure Gauge Calibration

Reference: ASME Section III, Article NB-6413 Published 2007

Requirement All test gauges shall be calibrated against a standard dead weight tester or a calibrated master gauge. The gauges shall be calibrated before each test or series of tests. A series of tests is that group of tests using the same pressure test gauge or gauges, which is conducted at the same site within a period not exceeding 2 weeks.

Observation: The test gages used to verify compliance for the hydrostatic test of the canister lid weld were required to be calibrated within 2 weeks of use. This requirement was stated in prerequisite Step 5.10 which required the pressure gauges for vent and drain ports to be calibrated within 2 weeks of the next test.

Documents Reviewed: a) Holtec Procedure, HPP-2253-300 "MPC Sealing at Callaway," Revision 7

Category: Pressure Testing **Topic:** Pressure Gauge Installation

Reference: ASME Section III, Article NB-6411 Published 2007

Requirement Pressure test gauges shall be connected directly to the component and visible to the operator controlling test pressure.

Observation: Pressure testing of the lid to shell weld at Callaway complied with the ASME requirements. Holtec Procedure HPP-2253-300 Attachment 8.10 "Hydrostatic Test System and Blowdown Setup," required the pressure gages to be directly connected to the Remote Valve Operating Assembly (RVOA), which connects to the vent and drain ports of the MPC. The hydrostatic pump was located next to the MPC in the Cask Washdown Pit making the test gages visible to the operator controlling the test pressure.

Documents Reviewed: a) Holtec Procedure HPP-2253-300, "MPC Sealing at Callaway," Revision 7

Category: Pressure Testing **Topic:** Pressure Gauge Ranges

Reference: ASME Section III, Article NB-6412 Published 2007

Requirement Analog type indicating pressure gauges used in testing shall be graduated over a range not less than 1.5 times nor more than 4 times the test pressure. Digital type pressure gauges may be used without range restriction.

Observation: Only digital pressure gauges were in use at Callaway during the preoperational dry-runs and the initial loading campaign for the Holtec UMAX ISFSI. Therefore, the gauges used at Callaway did not have use restrictions.

Documents Reviewed: a) Holtec Procedures HPP-2253-300, "MPC Sealing at Callaway," Revision 7

Category: Pressure Testing **Topic:** Thermal Expansion

Reference: ASME Section III, Article NB-6126 Published 2007

Requirement If a pressure test is to be maintained for a period of time and the test medium in the system is subject to thermal expansion, precautions shall be taken to avoid excessive pressure.

Observation: Precautions were taken to guard against over-pressurization of the canister during MPC sealing operations at Callaway. The Holtec procedure used at Callaway directed the technician to actively monitor pressure at all times during the hydro-static testing of the MPC lid to shell weld. Procedure steps 7.4.28 through 7.4.42 direct the technician to actively throttle the pressure while maintaining it within a certain acceptable band (125.5 – 129.5 psi) during the pressure test. NRC inspectors interviewed the Holtec cask loading supervisor and loading technician on the technique that would be followed. They both responded that the technician would be holding a throttle valve for the entire pressure test and maintaining the pressure manually. At any point during the test, the technician can release the pressure. The diligence of the technician would thereby limit the pressure to acceptable levels. There was also one relief valves in the line-up during the hydrostatic test, rated at 140 psi.

Documents Reviewed: a) Holtec Procedures HPP-2253-300, "MPC Sealing at Callaway," Revision 7

Category: Quality Assurance **Topic:** Approved QA Program

Reference: 10 CFR 72.140(d) Published 2015

Requirement A QA program previously approved by the Commission as satisfying the requirements of Appendix B to Part 50 will be accepted as satisfying the requirements of Part 72. In filing the description of the QA program required by Part 72.140(c), each licensee shall notify the NRC of it's intent to apply it's previously approved QA program to ISFSI activities. The notification shall identify the previously approved QA program by date of submittal, docket number and date of Commission approval.

Observation: The licensee had incorporated the Part 72 quality assurance requirements into their approved Part 50 quality assurance plan. Callaway sent a letter to the NRC on February 13, 2015, notifying the Commission of their intent to apply the previously approved 10 CFR Part 50 Quality Assurance Program to cover ISFSI activities at their site. Callaway's Operational Quality Assurance Manual (OQAM) Revision 31, was reviewed by inspectors during the programs review week and was found to adequately incorporate the Part 72 ISFSI activities.

Documents Reviewed: a) Callaway's Operational Quality Assurance Manual, Revision 31; b) Letter to NRC "Notification Pursuant to 10 CFR 72.140 (d) of Intent to Apply Previously Approved Quality Assurance Program to the ISFSI at Callaway Plant, Unit 1," dated 02/13/2015

Category: Quality Assurance **Topic:** Corrective Actions

Reference: 10 CFR 72.172 Published 2015

Requirement The licensee shall establish measures to ensure that conditions adverse to quality, such as failures, malfunctions, deficiencies, deviations, defective material and equipment, and nonconformances are promptly identified and corrected. In the case of significant conditions adverse to quality, the measures must ensure that the cause of the condition is determined and corrective action taken to preclude repetition. This must be documented and reported to appropriate levels of management.

Observation: The licensee had incorporated the tracking and addressing conditions adverse to quality of ISFSI activities into their Part 50 approved Corrective Action Program. Callaway Procedure APA-ZZ-00500 described the corrective action process and provided instructions for initiating a Corrective Action Request System (CARS). Plant management was involved in the process of reviewing CARs and periodic reports on the status of open or closed CARs.

Condition reports that had been issued related to the ISFSI activities and the fuel building crane were reviewed during the dry run and first loading inspections to evaluate whether conditions adverse to quality were being appropriately identified and adequately corrected. A large number of CARs written dealing with ISFSI construction, dry runs, programs review, and issues discovered during the first loading were reviewed by the NRC inspectors. The CARs were related to a variety of issues. The CARs reviewed were well documented and properly categorized based on the safety significance of the issue. The corrective actions taken were appropriate for the situations. Based on the comprehensiveness of the corrective action reports, the licensee demonstrated a high attention to detail in regard to the setup, maintenance, and operation of their ISFSI program and the cask handling crane. No NRC safety concerns were identified related to

the condition reports reviewed.

Documents Reviewed: a) Callaway Procedure APA-ZZ-00500, "Corrective Action Program," Revision 61; b) CARS # 201505151,2015505142, 201504137, 201503680, 201502667, 201502190, 201501660, 201501252, 201501155, 201501154, 201501047, 201406571, 201405949, and 201405323

Category: Quality Assurance **Topic:** Important to Safety Components - Ancillaries

Reference: FSAR 1032, Table 9.2.1 Revision 3

Requirement Ancillary equipment shall be classified as important to safety (ITS) in accordance with FSAR Table 9.2.1.

Observation: All Important to Safety (ITS) classifications contained in Table 9.2.1 of the HI-STORM FW FSAR (referenced by the UMAX FSAR) were consistent with the licensee's safety classifications in Holtec Document ID: 2253-C2015-46R2 for ancillary equipment except for the MPC Lifting Slings. Table 9.2.1 classified the lifting slings as ITS Category A while the licensee classified the slings as ITS Category B. An evaluation was performed by Holtec for the lifting slings in Purchase Specification (PS) PS-1234 that provided the justification for the slings as ITS Category B due to an evaluation that demonstrated that a 25 foot MPC drop will not breech the MPC containment boundary. Therefore, failure of a lifting sling would not directly cause loss of containment integrity and classification of the lifting slings as ITS Category B was justified. In addition, the inspectors noted in the FSAR Table 9.2.1 under the MPC Lift Attachments section, it stated, in part, that the ITS classification of the lifting device attached to the attachments may be lower than the attachment itself, as determined site-specifically. The MPC Lift Attachments are classified as ITS Category A equipment.

Documents Reviewed: a) Holtec Document 2253-C2015-46R2, "Safety Classification Summary of All Equipment to be Delivered Under Specification M-2020," Revision 1; b) Holtec Procedure PS-1234, "Purchase Specification for the MPC Downloading Sling for Downloading Using the VCT," Revision 4

Category: Quality Assurance **Topic:** Important to Safety Components - Cask System

Reference: FSAR 1032, Table 2.0.2, 2.0.3, 2.0.4, and 2.0.7 Revision 3

Requirement Structures, systems, and components of the HI-STORM UMAX cask system are identified as important to safety (ITS) in accordance with NUREG/CR-6407 "Classification of Transportation and Dry Spent Fuel Storage System Components." Holtec FW FSAR, Tables 2.0.2, 2.0.3, 2.0.4, and 2.0.7 provides a summary of the classification of the structures, systems, and components as important to safety A, B, C, and NTIS (not important to safety).

Observation: Callaway had listed the Important to Safety structures, systems, and components (SSCs) associated with the UMAX/FW systems appropriately in their Quality Assurance Program implementing procedures. Callaway Procedure APA-ZZ-00303 documented that the ITS SSCs associated with the UMAX were defined in Holtec Report HI-2135435. Callaway Procedure APA-ZZ-00303 Appendix 1, documented that the ITS SSCs associated with the MPC, HI-TRAC VW, and other loading equipment was

documented in Holtec Letter 2253-C2015-46R2. The Holtec report and letter were reviewed and were determined to be consistent with the tables in the FSAR and NUREG/CR-6407.

Documents Reviewed: a) Callaway Procedure APA-ZZ-00303, "Classification of Systems," Revision 16; b) Callaway Procedure APA-ZZ-00303, Appendix 1 "Callaway Director Olant System Classification Data," Revision 11; c) Holtec Report HI-2135435, "ITS Categorization for the UMAX System," Revision 2; d) Holtec Document ID:2253-C2015-46R2, "Letter from Holtec, Safety Classification Summary of All Equipment to be Delivered Under Specification M-2020," dated July 1, 2015

Category: Quality Assurance **Topic:** Instruments Requiring Calibration

Reference: FSAR 1032, Section 10.4 Revision 3

Requirement Instruments requiring calibration include flow rate monitors, canister pressure gauges, gas and water temperature gauges, temperature surface pyrometer, vacuum gauge for gas sampling and moisture monitoring instruments for Forced Helium Dehydration (FHD) operations.

Observation: Instruments requiring calibration were verified as being properly calibrated before use. NRC inspectors reviewed the calibration of many temperature gages, pressure gages, and flow meters during the dry run activities and the first loading. Callaway Procedure ETP-ZZ-04021 included requirements to verify the calibration of instruments consistent with the instruments listed in the Holtec FW FSAR table.

Documents Reviewed: a) Callaway Procedure ETP-ZZ-04021, "ISFSI Loading Campaign Performance - IPT," Revision 1

Category: Quality Assurance **Topic:** QA Audits

Reference: 10 CFR 72.176 Published 2015

Requirement The licensee shall carry out a comprehensive system of planned and periodic audits to verify compliance with all aspects of the QA program and to determine the effectiveness of the program.

Observation: The licensee had developed a comprehensive plan for auditing the spent fuel dry cask storage program which was described in Appendix B of Callaway's Operational Quality Assurance Manual, Revision 31. Callaway's quality assurance organization was conducting audits and surveillances of the dry cask storage activities at Callaway and of the cask vendor, Holtec, which included engineering design activities and the cask manufacturing activities.

Selected QA audits and surveillances were reviewed related to the Callaway's dry cask storage program. The documents reviewed included audits and surveillances of Holtec activities, including work being performed at the Holtec Manufacturing Division, as well as activities being conducted by both Holtec and the licensee's staff at the Callaway site. Audit and surveillance findings were adequately categorized, resolved and documented

Documents Reviewed: a) Callaway's Operational Quality Assurance Manual, Revision 31; b) Callaway Audit Report AP15009, "Dry Cask Storage System, Special Audit, AP15009," dated

05/08/2015

Category: Quality Assurance **Topic:** Receipt Inspection Checklists

Reference: FSAR 1032 Tables 9.2.4, 9.2.5; 1040 Table 10.1.1 Revision 3; 2

Requirement Holtec HI-STORM FW FSAR 1032 Tables 9.2.4 and 9.2.5 provide sample receipt inspection checklists for the canister and HI-TRAC transfer cask. Holtec HI-STORM UMAX FSAR 1040 Tables 10.1.1 a provide sample receipt inspection checklists for the VVM components. Users shall develop site-specific receipt inspection checklists.

Observation: The licensee had incorporated the UMAX FSAR requirements into Procedure ETP-ZZ-04021 and had performed the required receipt inspections for the multipurpose canister, HI-TRAC VW transfer cask, and VVM components. Procedure ETP-ZZ-04021 contained all the receipt inspection attributes listed in FSAR 1032 Table 9.2.4, MPC Inspection Checklist; Table 9.2.5, HI-TRAC VW Transfer Cask Inspection Checklist; and FSAR 1040 Table 10.1.1, HI-STORM UMAX VVM Assembly Inspection and Test Acceptance Criteria.

Documents Reviewed: a) Callaway Procedure ETP-ZZ-04021, "ISFSI Loading Campaign Performance - IPTE ," Revision 0

Category: Radiation Protection **Topic:** ALARA Program

Reference: FSAR 1040, Section 11.1.1 Revision 2

Requirement Licensees using the HI-STORM UMAX cask system will utilize and apply their existing site ALARA policies, procedures and practices for ISFSI activities to ensure that personnel exposure requirements of 10 CFR 20 are met.

Observation: The licensee had expanded their existing ALARA program to apply to ISFSI operations. The radiation protection program document (APA-ZZ-01000) described the ALARA Program. Section 4.4, addressed dose to members of the Public. Section 4.5.1, provided Federal Occupational Limits Guidelines. Section 4.5.2, discussed limits for pregnant workers. Section 4.5.4, outlined Administrative Dose Limits and subparagraph (c) noted that minors may not receive an occupational dose. Routine radiological controls were implemented by means of existing Callaway radiological control procedures associated with operation of the 10 CFR 50 facility. The licensee had reviewed and revised existing procedures to ensure compliance with requirements of 10 CFR 20.

The licensee applied lessons learned from other sites and incorporated suggestions from numerous INPO Reports. Man-hour and Person-rem projections were made using data from the Diablo Canyon ISFSI campaign. The licensee committed to use of low dose waiting areas. Two minute drills were observed during dry runs. Radiation Work Permits were developed. Personnel were encouraged to present ideas for reducing exposure during various evolutions. Radiation protection personnel were trained on the limitations of their survey instrumentation. The radiation protection technicians had participated in the Callaway Energy Site dry run demonstrations performed to meet Certificate of Compliance requirements. Radiation Protection personnel were found to be knowledgeable of the various activities and of the associated dose rates that would be expected. Health physics controls to include use of low dose waiting areas were

implemented during dry run and loading activities.

Documents Reviewed: a) Callaway Procedure APA-ZZ-01000, "Radiation Protection Program," Revision 41 b) General and Specific Radiation Work Permits as follows by prime number and subtitles - 15001753, "TRNSPRT – Move HI-TRAC From Wash down Pit to HI-STORM UMAX;" "CLOSE – Decontaminate, Close and Prepare MPC for Movement of Canister;" "LOAD – Place MPC into Spent Fuel Pool, Load Assemblies and Prepare for Movement to Cask Wash down Decontamination Pit;" "PREPS – Prepare the HiTrac and MPC for Loading"

Category: Radiation Protection **Topic:** Controlled Area Boundary Dose Rate Analysis

Reference: CoC 1040, Tech Spec A.5.3.2 Amendment 0

Requirement Considering the planned number of casks to be deployed and the cask contents, the licensee shall perform an analysis to confirm the dose limits of 10 CFR 72.104(a) will be satisfied under actual site conditions. 10 CFR 72.104(a) states that the annual dose to any real individual located beyond the controlled area must not exceed 25 mrem to the whole body, 75 mrem to the thyroid, and 25 mrem to any other critical organ as a result of direct radiation from the ISFSI during normal operations and anticipated occurrences. The results of the analysis shall be documented in the 10 CFR 72.212 evaluation report.

Observation: Callaway calculations provided in HPCI 15-05, noted that the annual dose to the nearest resident is 1.09E- 02 mrem, well below the 10 CFR 72.104(a) limit of 25 mrem. Holtec Calculation HI-2135879 in Section 5.3.1.2 compared the dose due to normal operations and anticipated occurrences due to the ISFSI to be compliant with limits of 10 CFR 72.104. Table 5.3-2 of the document noted an annual dose at 1600 meters, the approximate site boundary distance, and 2800 meters, the approximate distance to the nearest residence to be a maximum of less than 2 E-02 mrem, well within the limit of 25 mrem. The calculation assumed a maximum number of stored casks, 48. During loading the licensee measured the Transfer Cask and VVM surface neutron and gamma dose rated for comparison with limits and found results to be satisfactory. Because of the design of the welded and sealed canisters, there were no effluent pathways associated with the stored canisters under normal conditions.

The licensee was required by 10 CFR 72.104 to include doses from other nearby fuel cycle activities into the dose calculations. The operating Callaway Energy Center nuclear power plant as a source term was considered in the determination of whether the dose rate was being met. The license reviewed data for 2012, 2013 and 2014. While the data varied somewhat from year to year, the licensee determined that dose rates at the controlled area boundary were not statistically different from background levels.

Documents Reviewed: a) Callaway Procedure HSP-ZZ-0015, "Callaway Site Boundary Dose Evaluation," Revision 0; b) Callaway Calculation HPCI 15-05, "Evaluation of Direct Radiation Dose to the Member of the Public from the Independent Spent Fuel Storage Installation," Revision 1; c) Holtec Calculation HI-2135879, "ISFSI Site Boundary and CoC Dose Rate Calculations for Callaway Plant Site Boundary," Revision 0

Category: Radiation Protection **Topic:** Controlled Area Radiological Doses

Reference: 10 CFR 72.106(a)/(b)/(c) Published 2015

Requirement For each ISFSI, a controlled area must be established. Any individual located on or beyond the nearest boundary of the controlled area may not receive from any design basis accident 5 rem TEDE for accident conditions. Minimum distance from ISFSI to nearest boundary of controlled area must be 100 meters. Controlled area may include roads, railroads or waterways as long as arrangements are made to control traffic and protect public.

Observation: The ISFSI pad was located within the Callaway Energy Center nuclear power plant exclusion zone within the plant's owner controlled area. Holtec Calculation HI-213879 stated that the distance from the ISFSI to the nearest Callaway Plant site boundary is approximately 1600 meters which exceeds the 100 meter required. This calculation presented dose rates for the individual dose rate components (neutron and gamma) at the site boundary (2080 hours/year occupancy) and nearest residence (8760 hours/year occupancy). Results are a fraction of a mrem and well below regulatory limits of 25 mrem. Section 5.0 of Callaway's 72.212 Report stated that the MPC was designed to provide confinement of all radionuclides under normal, off-normal and accident conditions, including natural phenomena. Radioactive materials stored inside the MPC will not escape to the atmosphere over the life of the ISFSI. Therefore the dose rates expected to an individual was anticipated to be well below the regulatory limits.

Documents Reviewed: a) 10 CFR 72.212 Evaluation Report, "Callaway Plant, Unit 1, Dry Fuel Storage System for Spent Nuclear Fuel Docket 72-1045," Revision 0; b) Holtec Calculation HI-213879, "ISFIS Site Boundary and CoC Dose Rate Calculations for the Dry Storage Project," Revision 0

Category: Radiation Protection **Topic:** Dose Rate Survey - Transfer Cask

Reference: CoC 1040, TS A.5.3.3, A.5.3.4 (b) and A.5.3.8.(c) Amendment 0

Requirement The licensee shall establish site specific dose rate limits (gamma and neutron) for the sides of the transfer cask. The licensee shall measure the surface dose rates (gamma + neutron) for each loaded transfer cask. A minimum of 4 dose rate measurements shall taken on the side of the transfer cask at mid-plane, approximately 90 degrees apart around the circumference, and between the radial ribs of the water jacket. The measured dose rate shall not exceed the licensee's site-specific surface dose rate limits as determined from Technical Specification A.5.3.3 or 3500 mrem/hr on the side, whichever is lower.

Observation: Callaway Procedure HDP-ZZ-03000, Step 6.1.1.a.1 listed a maximum average dose rate of 3,500 mrem/hr, neutron and gamma combined, on the side of the HI-TRAC VW transfer cask. Procedure Step 6.1.2.a.1 required that HI-TRAC dose rate measurements be taken at mid-height plane locations approximately 90-degrees apart on the side of the HI-TRAC between the radial ribs of the water jacket. Those procedure steps aligned with the Holtec Certificate of Compliance Technical Specification requirements of A.5.3.3, A.5.3.4 (b), and A.5.3.8(c). NRC inspectors verified this criterion was met through procedure review.

Documents Reviewed: a) Callaway Procedure HDP-ZZ-03000, Appendix H, "Spent Fuel Storage Cask Surveys," Revision 0

Category: Radiation Protection **Topic:** Neutron Dosimetry

Reference: N/A

Requirement Neutron dose rates to occupational workers should be adequately monitored.

Observation: The licensee had incorporated provisions into the health physics monitoring program to adequately monitor for both thermal and higher energy neutron spectra. The licensee used Eberline NRD Instruments (rem balls) for field neutron dose rate surveys. Mirion DMC 2000GN electronic dosimeters provided for secondary dose monitoring. The dose of record was based on Landauer Optically Stimulated Luminescent (OSL) dosimeters combined with a Neutrak 144 Dual Element CR-39 system. The licensee's calculation and analysis presented results for various dosimeters and calibration spectra related to the HAWK Tissue Equivalent Proportional Chamber (TEPC), the Mirion DMPC 2000GN Electronic Dosimeter, and the Eberline NRD Neutron Dose Rate Survey Meter. The HAWK TEPC served as a reference for true neutron dose rate. This instrument allowed for the application of appropriate International Commission Radiation Protection (ICRP) quality factors to each interaction resulting in a more accurate response. The licensee committed to continued studies during campaigns to further refine correction and calibration factors and validate assumptions and decisions made.

Documents Reviewed: a) Callaway Calculation HPCI 08-02, "Response of Callaway Neutron Dosimeters and Neutron Survey Instruments," Revision 0

Category: Radiation Protection **Topic:** Shielding Effectiveness Test

Reference: FSAR 1040, Section 10.3.ii. Revision 2

Requirement Operational neutron and gamma shielding effectiveness tests shall be performed after the first fuel loading at the host plant site using written and approved procedures. Calibrated neutron and gamma dose rate meters shall be used to measure the actual neutron and gamma dose rates at the accessible surface of the HI-STORM UMAX VVM. The test is performed to identify the expected dose levels around the VVM in order to plan for appropriate radiation protection measures for future cask loadings.

Observation: Callaway had placed the shielding effectiveness test into their site Procedure HDP-ZZ-03000, Appendix H. Procedure HDP-ZZ-03000, Appendix H, Step 6.1.2.b. contained requirements to perform a radiation survey at multiple locations around the VVM closure lid and its outlet vent ducts. Step 6.1.4.b contained instructions to evaluate the VVM survey results to determine if 10 CFR 72.104 limits would be exceeded and if appropriate radiation controls were adequate.

Documents Reviewed: a) Callaway Procedure HDP-ZZ-03000 Appendix H, "Spent Fuel Storage Cask Surveys," Revision 0

Category: Radiation Protection **Topic:** Site-Specific Dose Rate Limits - Storage Cask

Reference: CoC 1040, Tech Spec A.5.3.3 (a), A.5.3.4 (a) Amendment 0

Requirement The licensee shall establish site-specific surface dose rate limits (gamma + neutron) for the top of the VVM. The surface dose rate limits for the storage cask may be set between the dose rate assumed in the 72.104(a) analysis and the dose rate needed to exceed the 72.104(a) dose limits, but shall not be set greater than 30 mrem/hr on the top of the VVM.

Observation: Callaway Procedure HDP-22-03000, Section 6.1.2.b.1.a requires that Tech Spec measurements be made at 4 locations taken against the outlet vent duct screen. The limits are listed in Section 6.1.1.b, combined neutron and gamma dose rates of 30 mrem/hr. Those procedure steps align with the Holtec certificate of compliance technical specification requirements of A.5.3.3 (a) and A.5.3.4 (a). NRC inspectors verified this criterion was met through procedure review.

Documents Reviewed: a) Callaway Procedure HDP-ZZ-03000, Appendix H, "Spent Fuel Storage Cask Surveys," Revision 0

Category: Radiation Protection **Topic:** Transfer Cask Surface Contamination Limit

Reference: CoC 1040, Tech Spec A.3.2.1 Amendment 0

Requirement Removable contamination on the exterior surfaces of the transfer cask and accessible portions of the canister shall not exceed 1,000 disintegrations per minute per 100 square centimeters (dpm/100 square centimeters) from beta and gamma sources and 20 dpm/100 square centimeters from alpha sources. The accessible portion of the canister is the upper portion of the canister external shell wall accessible after the inflatable annulus seal is removed and before the annulus shield ring is installed.

Observation: The contamination limits from TS A.3.2.1 of CoC 1040 had been incorporated into the licensee's procedures. Procedure HPP-2253-200, Step 7.9.11 and 7.9.14 required radiation protection group to perform contamination survey of MPC top lid surfaces and accessible areas on the MPC after the annulus seal was removed to ensure the TS 3.2.1 had been met.

The requirements in the Callaway procedures were consistent with the Holtec FW FSAR, Section 9.2.4.2, which stated that after decontaminating the canister lid top and the shell area above the annulus seal, to deflate the seal and survey the canister lid top surface and the accessible areas of the top three inches of the canister. A "Note" preceding Step 7.9.14 in HPP-2253-200 stated: "The MPC exterior shell survey is performed. Indications of contamination could require the MPC to be unloaded. In the event that the MPC shell is contaminated, users must decontaminate the annulus. If the contamination cannot be reduced to acceptable levels, the MPC must be returned to the spent fuel pool and unloaded. The MPC may then be removed and the external shell decontainінated."

Documents Reviewed: a) Holtec Procedure HPP-2253-200, "MPC Loading at Callaway," Revision 9

Category: Records **Topic:** Cask Records

Reference: 10 CFR 72.234(d)(2) & (d)(3) Published 2015

Requirement A list of records required for each cask is provided in 10 CFR 72.234(d)(2). The certificate holder is required by 10 CFR 72.234(d)(3) to provide an original of these records to the user.

Observation: The licensee was maintaining the required records in their quality related records system consistent with 10 CFR 72.234. Callaway had implemented retention of ISFSI records into their Part 50 record retention program. Callaway Procedure APA-ZZ-00209 Step 7.18.1.h. required retention of all Quality Assurance records pertaining to Callaway ISFSI Important to Safety structures, systems, and components per 10 CFR 72.174. NRC inspectors reviewed the cask records provided to Callaway per 10 CFR 72.234(d)(2) from Holtec for the first canister planned to be loaded by Callaway. The canister package contained the required information and was easily retrieved through Callaway's record retention program.

Documents Reviewed: a) Callaway APA-ZZ-00209, "Records Identification, Retention and Destruction," Revision 18

Category: Records **Topic:** Notice of Initial Loading

Reference: 10 CFR 72.212(b)(1) Published 2015

Requirement The general licensee shall notify the NRC at least 90-days prior to first storage of spent fuel.

Observation: Callaway notified the NRC by letter dated January 27, 2015, of the plans to begin fuel loading at the Callaway site on or after April 27, 2015. This notification met the requirements of the 90 day notification of initial loading required by 10 CFR 72.212(b)(1).

Documents Reviewed: a) Letter (ULNRC-06163) from David W Neterer, Vice President - Nuclear Operations Ameren Missouri, entitled "Docket Number 50-483 and 72-1045 Callaway Plant Unit 1, Union Electric Co., Facility Operating License NPF-30, 90-Day Notification Pursuant to 10 CFR 72.212(b)(1) of Intent to Load Spent Fuel Under a General License," dated January 27, 2015

Category: Records **Topic:** Record Retention for 72.212 Analysis

Reference: 10 CFR 72.212(b)(5)(iii) Published 2015

Requirement A copy of the 10 CFR 72.212 analysis shall be retained until spent fuel is no longer stored under the general license issued under 10 CFR 72.210.

Observation: Callaway had implemented retention of ISFSI records including the 10 CFR 72.212 Evaluation Report into their Part 50 record retention program. Callaway Procedure APA-ZZ-00209 Step 7.18.1.h. required retention of all Quality Assurance records pertaining to Callaway ISFSI Important to Safety structures, systems, and components per 10 CFR 72.174. Callaway generated a print-out copy of File Plan E430.0002 that documented the ISFSI records shall be retained for additional 5 years after the fuel has been transfer in accordance with 10 CFR 72.72 (d).

Documents Reviewed: a) Callaway Procedure APA-ZZ-00209, "Records Identification, Retention and Destruction," Revision 18; b) File Plans E430.0002, "ISFSI Records Retained for Department of Energy Transfer of Spent Fuel," dated 05/12/15

Category: Records **Topic:** Registration of Casks with NRC

Reference: 10 CFR 72.212(b)(2) Published 2015

Requirement The general licensee shall register the use of each cask with the NRC no later than 30 days after using the cask to store spent fuel.

Observation: The requirement to notify the NRC within 30 days of using a cask to store spent fuel was incorporated into Procedure ETP-ZZ-04021. Step 7.7.8 of the procedure, required the licensee to ensure loaded MPCs have been registered with the NRC no later than 30 days after using the cask to store fuel.

Documents Reviewed: a) Callaway Procedure ETP-ZZ-04021, "ISFSI Loading Campaign Performance IPTE," Revision 1

Category: Safety Reviews **Topic:** Changes, Tests, and Experiments

Reference: 10 CFR 72.48(c)(1) Published 2015

Requirement A licensee can make changes to their facility or storage cask design if certain criteria are met as listed in 10 CFR 72.48.

Observation: The licensee had combined the 72.48 screening and evaluation process with the 50.59 process used at the site. Procedure APA-ZZ-00143 described the screening and evaluation process for both requirements and used several different forms.

The licensee had developed classroom training material and an engineering qualification card for the 10 CFR 72.48 program. The training material provided a good description of the purpose and philosophy of the 10 CFR 72.48 process and how it related to other processes that controlled licensing basis activities at Callaway. Relevant definitions and applicable terms were provided with discussions of what they meant. The training material was well developed and very informative in assisting the user in making decisions related to the 72.48 screening and evaluation process. Examples were provided to further illustrate how to implement the process. The training material included numerous drawings and pictures describing the various components of the dry cask system specific to Callaway to help familiarize the personnel assigned to perform the screenings and evaluations with the key safety components of the various systems. The training module described the forms required and gave examples of what the various screening and evaluation criteria meant. Lessons learned at other sites relevant to the 10 CFR 72.48 process were provided.

As part of the 72.212 Evaluation Report review, NRC inspectors reviewed many 72.48 screens and three full 72.48 evaluations that related to the sites fire hazards analysis, explosion hazards, and the tornado missile analysis. The full evaluations were documented in 72.48 Evaluation Log No 15-01.

The HI-STORM UMAX FSAR documented that the fire accidents for storage were

conservatively postulated to be the results of the spillage and ignition of 50 gallons of combustible transporter fuel. The Callaway site specific analysis evaluated a fire due to the Low Profile Transporter (HI-PORT), tracked Vertical Cask Transporter (VCT), and an Arial Work Platform (JLG) boom lift. The fire specific analysis was documented in Holtec Report HI-2156590. The fire accident was evaluated using the same methodology as in the FSAR. The results of the analysis concluded that the combined combustible liquids from the above listed equipment would not cause fuel assemblies to exceed peak cladding temperatures above the FSAR allowable accident limits of 570 degrees C.

The explosion hazard documented in the FSAR, found the UMAX was qualified to a 10 psi overpressure. Due to the location of the ISFSI within the Protected Area, Callaway requested that the UMAX be reviewed and qualified for an overpressure of 20 psi. The explosion impacts from Callaway were documented in Holtec Report HI-2146196. The Callaway site-specific explosion hazards were evaluated and determined that the overpressure wave did not result in lid separation and that all lid stresses were a fraction of the allowable limits.

The tornado missile analysis contained in the FSAR did not bound all the missiles contained in the Callaway's Part 50 UFSAR Table 3.5-1. As a result, a site specific analysis was performed using a different set of missiles to bound Callaway's design basis missiles. The tornado missile analysis was documented in Holtec Report HI-2146196. The specific analysis utilized the same methodology as performed in the Holtec UMAX FSAR. The kinetic energy generated by the Callaway Part 50 design basis missiles was bounded by the kinetic energy input values generated from the new list of bounding missiles that were presented in the analysis. It was found that the new bounding missiles did not breach the confinement boundary, locally deform the cask such that the retrievability of the MPC was threatened, or deform the cask plastically such that the shielding effectiveness was affected.

All three evaluations were performed using the original methodology and acceptance criteria, and found that the original acceptance criteria was still met. The evaluations documented that the site specific conditions did not result in more than a minimal increase in the frequency or likelihood of any accident or malfunction, or consequences of an accident or malfunction. The site specific evaluations also did not create the possibility of a different type of accident or malfunction, cause a design basis limit for a fission product barrier to be exceeded or altered, or result in a change in methodology. Therefore the 72.48 evaluation concluded the activity can be implemented without requesting a CoC Amendment.

Documents Reviewed: a) Callaway Procedure APA-ZZ-00143, "10 CFR 50.59 and 10 CFR 72.48 Reviews," Revision 16; b) Training Course Number T62.06496, "10 CFR 72.48 "Evaluator Initial Training," Date 01/17//2015; c) Callaway Engineering Qualification Standard, "Prepare a 10 CFR 72.48 Screening and Evaluation," Date 12/10/14; d) 72.48 Evaluation Log No. 15-01, "MP 14-0014, Dry Fuel Storage Licensing and Operations Documentation," Revision 0; e) Holtec Report HI-2146196, "Evaluation of Plant Hazards at Callaway Energy Center," Revisions 2, 3, and 4; f) Holtec Report HI-2156590, "Evaluation of Combined Effect of HI-PORT and VCT Fires on HI-TRAC at Callaway," Revisions 0

and 1; g) Holtec Report HI-2135677, "Evaluation of Effects of Tracked VCT Fire on HI-STORM FW System," Revision 5; h) 10 CFR 72.212 Evaluation Report, "Callaway Plant, Unit 1, Dry Fuel Storage System for Spent Nuclear Fuel Docket 72-1045," Revision 0

Category: Slings **Topic:** Sling Heavy Load Requirements

Reference: NUREG 0612, Section 5.1.6 (1) (b) Published July 1980

Requirement Dual or redundant slings should be used such that a single component failure or malfunction in the sling will not result in an uncontrolled lowering of the load, OR the load rating of the sling should be twice the sum of the static and dynamic loads.

Observation: Dual and redundant slings were used to download the canister from the transfer cask into the HI-STORM UMAX. The load rating on the slings was twice the weight of the static and dynamic loads of the canister. The slings purchased at Callaway had a vertical rating of 140,000 pounds each. Two of these slings are required for downloading the canister. The fully loaded canister was calculated to be 97,288 pounds per Holtec Document HI-2146011 (Case 6). A conservative static value of 120,000 pounds was used in the calculation of Purchase Specification PS-1234 for purchase of the slings. A dynamic load of 15% was added to the static load providing a total of 140,000 pounds. Redundant slings were used in the vertical formation. The minimum vertical rated capacity of each sling was calculated to be 140,000 pounds. The slings that were purchased by Callaway were rated at 140,000 lb (vertical rated) capacity (TPSE-EE14000 - 62.5ft Twin-Path Spark-eater eye & eye slings).

NRC inspectors reviewed the purchase specifications for the slings utilized in lifting the MPC lid, lifting the gate adapter, lifting the closure lid, and other miscellaneous slings utilized throughout the campaign. Dual and redundant slings were used in all heavy load lifts associated with the loading operations.

Documents Reviewed: a) Holtec Report HI-2146011, "Cask Handling Weights at Callaway," Revision 1; b) Holtec PS-1234, "Purchase Specification for the MPC Downloading Slings for Downloading Using the VCT," Revision 4; c) Holtec PS-3200, "Purchase Specification for the HI-STORM FW System Multi-Purpose rigging system," Revision 3

Category: Slings **Topic:** Sling Identification

Reference: ASME B30.9, Section 9-5.1.6 Published 1990

Requirement Each sling should be permanently marked to show: (a) name or trademark of manufacturer; (b) manufacturer's code or stock number; (c) rated loads (rated capacities) for the types of itches used; (d) type of natural or synthetic material;

Observation: NRC inspectors visually inspected slings during the outside pad operations dry-run, from July 14-17, 2015. All inspected steel wire rope and synthetic fiber slings were tagged with information including manufacturer, rated load capacity, serial number, and type of material.

Documents Reviewed: N/A.

Category: Slings **Topic:** Sling Inspections - Frequent

Reference: ASME B30.9, Section 9-4.7.1 (b) Published 1990

Requirement A visual inspection for damage shall be performed each day or shift the sling is used.

Observation: Callaway's rigging procedure contains the routine inspection criteria for wire rope and synthetic slings. According to the procedure, slings were to be inspected for all manner of wear and tear, including burns, missing tags, holes, cuts, etc on a daily basis. NRC inspectors noted that slings were inspected during dry-run activities and during the initial cask loading operations at Callaway. The licensee had established training and procedures to ensure that slings would be properly inspected prior to use during fuel loading operations for the Callaway ISFSI.

Documents Reviewed: a) Holtec Report No.: HI-2135598, "MPC Lift Sling Operations and Maintenance Manual," Revision 0; b) Callaway Procedure APA-ZZ-00365, Addendum R, "Callaway Rigging Operations," and Attachment 4, "Portable Hoist and Come-along Frequent Inspections," Revision 1

Category: Slings **Topic:** Sling Inspections - Periodic

Reference: ASME B30.9, Section 9-4.7.1 Published 1990

Requirement A complete inspection for damage to the sling shall be conducted at intervals not to exceed one year.

Observation: A complete inspection of slings was completed on an annual basis for the slings utilized in the Callaway loading campaign. All slings and wire ropes utilized in the Callaway loading operations were owned and controlled by Holtec. Holtec Procedure HSP-410 was utilized to perform the annual inspections. The NRC inspector confirmed that all slings used during the first canister loading operations were within their annual inspection dates by visual examination of sling equipment identification tags.

Documents Reviewed: a) Holtec Procedure HSP-410, "NUREG-0612 Periodic Maintenance Program: Control of Slings, Hooks, Misc. Tackle and Structural and Mechanical Lifting Devices," Revision 0

Category: Slings **Topic:** Sling Load Rating

Reference: NUREG 0612, Section 5.1.1 (5) Published July 1980

Requirement In selecting the proper sling, the load used should be the sum of the static and maximum dynamic load. The rating identified on the sling should be in terms of the "static load" which produces the maximum static and dynamic load.

Observation: All slings utilized at Callaway were selected in accordance with the NUREG 0612 requirements. NRC inspectors reviewed the Purchase Specifications for the slings utilized in downloading the MPC, lifting the MPC lid, lifting the gate adapter, lifting the closure lid, and other misc. slings utilized throughout the campaign. The documents demonstrated the slings selected were based on the sum of the static and dynamic loads. NRC inspectors inspected the slings that were utilized in the campaign and confirmed that the slings met the Purchase Specifications.

Documents Reviewed: a) Holtec PS-1234, "Purchase Specification for the MPC Downloading Slings for Downloading Using the VCT," Revision 4; b) Holtec PS-3200, "Purchase Specification for the HI-STORM FW System Multi-Purpose rigging system," Revision 3

Category: Slings **Topic:** Sling Proof Loading

Reference: ASME B30.9, Section 9-5.4 Published 1990

Requirement When specified by the purchaser, slings of all types shall be proof loaded. The proof load for single leg (branch) slings and endless slings shall be two times the vertical rated load (rated capacity).

Observation: Holtec Purchase Specification, Section 8, "Inspections and Testing Requirements," required that a proof test of twice the rated vertical capacity be applied to all load bearing components of the MPC downloading ancillary. The certificate of conformance from I&I Slings showed that two slings, serial numbers P102114094 and P102114095, were proof tested to twice their rated load of 140,000 lbs., in accordance with ASME 30.9 standards. This criteria was satisfactorily demonstrated by documents review by NRC inspectors during dry-run activities and first loading at Callaway for its Holtec UMAX ISFSI.

Documents Reviewed: a) Holtec International Purchase Specification PS-1234, "Purchase Specification for the MPC Downloading Sling for Downloading using the VCT," Revision 4; b) Holtec International COC-14311-002, "Certificate of Conformance," Revision 0

Category: Slings **Topic:** Sling Temperature Limits

Reference: No Reference Provided N/A

Requirement Synthetic slings shall not be used in contact with objects that exceed the temperature limit of the sling.

Observation: NRC inspectors verified that a maximum allowable sling to MPC contact temperature had been established for cask loading operations at the Callaway ISFSI. The synthetic MPC lift slings used at Callaway had a specified maximum contact temperature for the synthetic sling of no more than 300 degrees F in both the Holtec Report and Holtec's procedure for use at Callaway.

Documents Reviewed: a) Holtec Report No.: HI-2135598, "MPC Lift Sling Operations and Maintenance Manual," Revision 0; b) Holtec Procedure HPP-2253-400, "MPC Transfer for Callaway," Revision 7

Category: Slings **Topic:** Synthetic Round Sling Removal from Service

Reference: ASME B30.9, Section 9-4.8 Published 1990

Requirement A synthetic round sling shall be removed from service if any of the following conditions are present: (a) cuts, gouges, badly abraded spots; (b) seriously worn surface fibers or yarns; (c) considerable filament or fiber breakage along the line where adjacent strands meet (light fuzzing is acceptable); (d) particles of broken filament or fibers inside the rope between the strands (inspect inside the rope); (e) discoloration or harshness that may mean chemical damage or excessive exposure to sunlight. Inspect filaments or

fibers for weakness or brittleness; (f) kinks or hockles; (g) melting or charring on any part of the sling; (h) excessive pitting or corrosion, or cracked, distorted or broken fittings; (i) other visible damage that causes doubt as to the strength of the sling

Observation: The synthetic round sling inspection criteria, consistent with the key elements of ASME B30.9, was listed in Procedure HSP-410. All slings and wire ropes utilized in the Callaway loading operations were owned and controlled by Holtec. The Holtec Procedure HSP-410 contained the ASME B30.9 removal criteria in Section 6.3.6. The NRC inspectors confirmed that all slings used during the first canister loading operations were within their annual inspection dates.

Documents Reviewed: a) Holtec Procedure HSP-410, "NUREG-0612 Periodic Maintenance Program: Control of Slings, Hooks, Misc. Tackle and Structural and Mechanical Lifting Devices," Revision 0

Category: Slings **Topic:** Wire Rope Sling Removal From Service

Reference: ASME B30.9, Section 9-2.8.3 Published 1990

Requirement A wire rope sling shall be removed from service if any of the following conditions are present: (a) for strand laid and single part slings ten randomly distributed broken wires in one rope lay, or five broken wires in one strand in one rope lay; (b) severe localized abrasion or scraping; (c) kinking, crushing, birdcaging or any other damage resulting in distortion of the rope structure; (d) evidence of heat damage; (e) end attachments that are cracked, deformed, or worn to the extent that the strength of the sling is substantially affected; (f) severe corrosion of the rope or end attachments;

Observation: The wire rope sling inspection criteria, consistent with the key elements of ASME B30.9, was listed in Procedure HSP-410. All slings and wire ropes utilized in the Callaway loading operations were owned and controlled by Holtec. The Holtec Procedure HSP-410 contained the ASME B30.9 removal criteria in Section 6.3.5. The NRC inspectors confirmed that all slings used during the first canister loading operations were within their annual inspection dates.

Documents Reviewed: a) Holtec Procedure HSP-410, "NUREG-0612 Periodic Maintenance Program: Control of Slings, Hooks, Misc. Tackle and Structural and Mechanical Lifting Devices," Revision 0

Category: Special Lifting Device **Topic:** Lifting Device's Annual Testing

Reference: ANSI N14.6, Sect 7.3.1; Sect 6.3.1 Published 1993

Requirement Annually, not to exceed 14 months, all special lifting devices shall be subjected to a test load equal to 300% of the maximum service load if a single component failure on the device could result in an uncontrolled lowering of the load. If the design for handling the load incorporates a single-failure proof concept, then each path in the dual-load-path device shall be tested to 150% of the load instead of the 300%. After sustaining the load for a period of not less than 10 minutes, critical areas, including major load bearing welds, shall be subject to visual inspection for defects and all components shall be inspected for permanent deformation. In cases where surface cleanliness and conditions permit, the load testing may be omitted and dimensional testing, visual inspection and

nondestructive testing of major load-carrying welds and critical areas shall suffice.

Observation: Callaway/Holtec equipment had recently performed the initial load tests and NDE examinations for the lift yoke, yoke extension, HI-TRAC VW lifting lugs, MPC lift cleats, and the VCT lift links for the current year. NRC inspectors reviewed two worksheets that were produced for recording/tracking future test results for the Callaway owned equipment. Both Callaway owned and Holtec owned equipment's' preventative/routine maintenance procedures had not been fully developed at the time of the initial loading inspection. This will be a follow-up item to be reviewed in future inspections at Callaway.

Documents Reviewed: a) Callaway Worksheets PM15506409, "Lift Yoke Annual Load Test," Revision 0; b) PM15506408, "Lift Yoke Extension Annual Load Test," Revision 0

Category: Special Lifting Device **Topic:** Lifting Device's Initial Acceptance Testing

Reference: ANSI N14.6, Sect 7.3.1; Sect 6.2.1; Sect 6.5 Published 1993

Requirement Prior to initial use, the yoke shall be subjected to a test load equal to 300% of the maximum service load if a single component failure on the yoke could result in an uncontrolled lowering of the load. If the design for handling the load incorporates a dual-load-path concept, then each path in the dual-load-path device shall be tested to 150% of the load instead of the 300%. After sustaining the load for a period of not less than 10 minutes, critical areas, including load bearing welds, shall be subject to nondestructive testing using liquid penetrant or magnetic particle examination.

Observation: NRC inspectors reviewed load test procedures and acceptance testing results for the Lift Yoke, S/N: 702-0872-8851-1000-1; Lift Yoke Extension, S/N: 0929-878+91000-1-A; Lift Lug S/Ns: 723-006 "G" and 723.-010 "H;" and the MPC Lift Cleats and Shielding, S/N: 209-98131000-1. Those load tests included the results of hydraulically simulated loads of 300% the maximum rated load capacity, as specified by Holtec Procedure HSP-706. The inspection criteria included sustaining the simulated load for a minimum of ten minutes followed by dimensional measurements, which were at specified locations, pre and post load test. The inspection also called for visual testing and non-destructive testing of the special lifting devices and those results were also shared with the NRC. The lift yoke, lift yoke extension, both lift lugs, and the MPC lift cleats all passed the load, dimensional, and nondestructive testing that was required by ANSI N 14.6.

Documents Reviewed: a) Holtec Purchase Specification PS-3702, "Purchase Specification for the HI-TRAC VW Lift Yoke," Revision 5; b) Holtec Procedure HSP-706, "HI-TRAC VW Lift Yoke/Lift Lugs Load Test Procedure," Revision 2; c) Holtec Procedure HPP-2253-06, "Lift Yoke Extension Factory Acceptance Test," Revision 0; d) Holtec PS-3701, "Purchase Specification for the HI-TRAC VW Lift Links," Revision 2; e) Holtec Report No: HI-2135597, "MPC Lift Cleats Operation and Maintenance Manual," Revision 0; f) Holtec Procedure HPP-2253-06, "Lift Yoke Factory Acceptance Test," dated January 20, 2015; g) Holtec Procedure HSP-710, "MPC FW Lift Cleat Load Test Procedure," Revision 4

Category: Special Lifting Device **Topic:** Lifting Device's Stress Design -Dual-Load-Path

Reference: ANSI N14.6, Sect 4.2.1.1; Sect 4.1.3; Sect 7.2.3 Published 1993

Requirement For yokes that are single failure proof by having dual-load-path attachments, the load bearing members of the yoke shall be capable of lifting three (3) times the combined weight of the shipping container plus the weight of the intervening components of the special lifting device, without generating a combined shear stress or maximum tensile stress at any point in the device in excess of the corresponding minimum tensile yield strength of the material of construction. They shall also be capable of lifting five (5) times the weight without exceeding the ultimate tensile strength of the materials. The dual load-path attachment points on the yoke shall be designed such that each load path will be able to support a static load of three (3) times the weight of the critical load, including intervening components of the lifting device.

Observation: The lift yoke, as well as, the yoke extension, the HI-TRAC VW Lifting Lugs, the MPC lift cleats, the VCT lift links, and the transporter's pulley system used to download the MPC were all designed in accordance with the ANSI N14.6 standard and NUREG 0612 requirements. Stress reports and design specifications for each lifting device were reviewed to confirm the ANSI and NUREG design requirements were properly specified. Additionally, the NRC inspectors reviewed the bill of materials specified on the design drawing or in the components' documentation packages to confirm the components were fabricated from the same material as specified in the purchase specifications and stress reports.

Documents Reviewed: a) Holtec DOC-104-702-037, "HI-TRAC VW Lift Yoke for Callaway," Revision 0; b) Holtec Drawing 2253-8851, "Assembly of Lift Yoke HI-TRAC VW," Revision 1; c) Holtec Drawing 104-8997, "HI-TRAC VW Lift Link," Revision 4; d) Holtec PS-1120, "Purchase Specification for the Vertical Cask Transporter," Revision 6; e) Holtec PS-3702, "Purchase Specification for the HI-TRAC VW Lift Yoke," Revision 5; f) Holtec Report HI-2146172, "Structural Analysis of HI-TRAC VW Lift Yoke," Revision 0; g) Holtec PS-3723, "Purchase Specification for the HI-TRAC VW Lifting Lugs," Revision 4; h) Holtec PS-3117, "Purchase Specification for the HI-TRAC VW Lift Yoke Extension," Revision 2; i) Holtec Report HI-2146002, "Structural Evaluation of Lift Yoke Extension at Callaway," Revision 1; j) Holtec Report HI-2135524, "Structural Analysis of the HI-TRAC VW Lift Links (250,000lbs Capacity)," Revision 1; k) Holtec PS-3701, "Purchase Specification for the HI-TRAC VW Lift Links," Revision 2; l) Holtec Report HI-2125307, "Structural Analysis of HI-TRAC VW Lift Lug," Revision 4

Category: Special Lifting Device **Topic:** Special Lifting Devices Inspection Prior to Use

Reference: ANSI N14.6, Sect 6.3.6 Published 1993

Requirement The yoke shall be visually inspected by operating personnel for indications of damage prior to each use.

Observation: Prior to use inspections steps were placed in Callaway's loading procedures prior to rigging any of the special lifting devices used during the ISFSI loading operations. This included the yoke, yoke extension, HI-TRAC VW lift lugs, MPC cleats, and VCT lift links.

Documents Reviewed: a) Procedure HPP-2253-200, "MPC Loading at Callaway," Revision 9; b) Procedure HPP-2253-400, "MPC Transfer at Callaway," Revision 7

Category: Special Lifting Device **Topic:** Transporter Initial Acceptance Testing

Reference: ANSI N14.6, Sect 7.3.1; Sect 6.2.1; Sect 6.5 Published 1993

Requirement Prior to initial use, the special lifting device shall be subjected to a test load equal to 300% of the maximum service load if a single component failure on the yoke could result in an uncontrolled lowering of the load. If the design for handling the load incorporates a single-failure proof concept, then each path in the dual-load-path device shall be tested to 150% of the load instead of the 300%. After sustaining the load for a period of not less than 10 minutes, critical areas, including load bearing welds, shall be subject to nondestructive testing using liquid penetrant or magnetic particle examination.

Observation: Review of the factory acceptance test documentation showed that the VCT used by Callaway for its initial ISFSI campaign had met all of the regulatory requirements. The Vertical Cask Transporter (VCT) used at Callaway for transfer of the MPC from the fuel building to the ISFSI pad had its initial acceptance testing performed on April 9 and 10, 2015. NRC inspectors reviewed the test records that documented the 125% static load test, 100% dynamic load and operational test, 150% MPC downloader system test, initial and post-test non-destructive examination (NDE), and others.

Documents Reviewed: a) Holtec Procedure HSP-199, "VCT Factory Acceptance Test Procedure," Attachment 7.1, "FAT [Factory Acceptance Test] and Sign-offs," Revision 2

Category: Special Lifting Device **Topic:** Transporter Inspection - Quarterly

Reference: ANSI N14.6, Sect 6.3.7 Published 1993

Requirement Special lifting devices shall be visually inspected by maintenance or other non-operating personnel at intervals not to exceed three months in length for indications of damage or deformation.

Observation: Callaway's Procedure ETP-ZZ-04021 had included acceptance criteria for vendor owned (Holtec) special lifting devices in Step 7.1.12 for the HI-TRAC lift lugs and Step 7.1.28 for the MPC lid lift cleats, which included making sure that all periodic maintenance and checks had been performed. This procedure also included the Callaway owned special lifting devices of the lift yoke and lift yoke extension in Step 7.1.19. NRC inspectors reviewed the quarterly preventative maintenance worksheets for both the lift yoke and lift yoke extension. NRC verified that all of the proper procedures and supporting documents were in place to support the quarterly visual inspection of special lifting devices or to check to insure that vendor supplied special lifting devices had been properly tested prior to use at Callaway.

Documents Reviewed: a) Callaway Procedure ETP-ZZ-04021, "ISFSI Loading Campaign Performance – IPTE (Infrequently Performed Tests and Evolutions)," Revision 0

Category: Storage Operations **Topic:** Storage Cask Temperature Monitoring

Reference: CoC 1040, Tech Spec A.3.1.2 Amendment 0

Requirement Verify all VVM outlet air ducts are free of blockage from solid debris or floodwater every 24 hours, OR for VVMs with installed temperature monitoring equipment, verify that the difference between the average VVM air outlet temperature and ISFSI ambient temperature is less than or equal to 80 degrees F every 24 hours for storage casks containing PWR canisters.

Observation: The licensee planned to conduct vent screen daily observations per operator rounds procedure ODP-ZZ-0016E, Appendix 1, Point ID OPS 02174. If debris was found, the operator would clear the debris from the vents. If the clearance could not be accomplished by hand, a work order would be created to clear the debris. Results were communicated to a licensed operator who documented the results in OSP-ZZ-00001, Attachment 4 and in the Operator Log. If there was a blockage, OSP-ZZ-00001 directed the operator to review the applicable Technical Specification for applicability.

Documents Reviewed: a) Callaway Procedure ODP-ZZ-0016E Appendix 1, "Inside Operator Rounds," dated 06/29/2015; b) Callaway Procedure OSP-ZZ-00001, "Control Room Shift and Daily Log Readings and Channel Checks," Attachment 4, Revision 85

Category: Storage Operations **Topic:** Thermal Acceptance Test

Reference: FSAR 1040 Section 10.3.iii Revision 2

Requirement A thermal acceptance test shall be performed on the first fully loaded VVM assembly whose aggregate MPC heat load is at least 50% of the Design Basis maximum heat load per the system CoC.

Observation: Callaway placed limits on their first campaign's canisters' heat loads to ensure all MPCs would have a heat load below 50% of the Design Basis maximum. By placing this restriction, Callaway would not be required to perform the Heat Transfer Validation Test. Callaway Procedure ETP-ZZ-04020 Section 5.1, noted that the aggregate heal load of the loaded elements will not exceed 33.88 KW which is 80 percent of the design basis limit of 43.5 kW. A note in the Section 5.1 listed 21.75 kW as the threshold for when a thermal performance test would be required.

Table 8-8H of Procedure ETP-ZZ-04020, provided decay heat load for selected elements including inserts and approved MPC Regions for loading. Attachment 1 to the procedure provided fuel assembly identification numbers, insert identification numbers, decay heat rate per assembly, assembly burnup, spent fuel pool location prior to loading operations, the cell location in the MPC into which the assembly is to be loaded, the decay heat limit for the specified MPC cell, and the MPC cell allowed inserts for the six canisters that were to be loaded in the first campaign.

The licensee completed a two party compliance checklist for each MPC loaded in the campaign. The checklist verified that all assemblies met the MPC cell decay heat limit, that the total heat load was less that 21.75 kW, the all assembly burnups were equal to or less than 68,200 MWD/MTU, that all fuel assemblies are Westinghouse 17X17 Zr clad assemblies, that all assemblies have been discharged for more than 3 years, that all fuel

assemblies are approved for the designated MPC region, the MPC contained no nuclear source assemblies, that RCCAs were only placed in cells 5 through 7, 10 through 14, 17 through 21, 24 through 28 and 31 through 33, that BRPA maximum burnup was 60,000 MWD/MTU and that RCCA and Thimble plug maximum burnup is 630,000 MWD/MTU. Review of the data supplied for each MPC loading indicated that applicable limits were met.

Documents Reviewed: a) Callaway Procedure ETP-ZZ-04020, "Fuel Selection and Cask Loading for Dry Cask Storage," Revision 0; b) Cask Loading Plan, Attachment 1 to ETP-ZZ-04020, Revision 0, for MPC# HGMPC0037,HGMPC0038, HGMPC0039, HGMP0040, HGMP0041, HGMP0042

Category: Storage Operations **Topic:** VVM Vent Screen Inspections

Reference: FSAR 1040, Table 10.4.1 Revision 2

Requirement The VVM vent screens shall be visually examined for damage monthly.

Observation: The licensee incorporated the FSAR requirement to perform monthly visual examinations of the VVM vent screens into the Preventative Maintenance system as PM # 1008329. This maintenance item required a monthly visual check of the VVM screens for damage and accessible VVM areas for long term degradation.

Documents Reviewed: a) Callaway PM #1008329, "VVM Monthly Inspections," Revision 0

Category: Unloading Operations **Topic:** Canister Gas Sampling

Reference: FSAR 1032, Section 9.4.3.4 Revision 3

Requirement During unloading of a cask, gas sampling is performed to assess the condition of the fuel assembly cladding. The gas sample bottle is connected to the vent port RVOA and the RVOA body and sample bottle are evacuated. The vent port cap is then slowly opened using the RVOA, and the gas sample is obtained.

Observation: Callaway demonstrated the ability to draw a gas sample from a canister during the fluid operations dry run that was conducted on June 2-4, 2015. Callaway would utilize Procedure CSP-ZZ-07046 to obtain a gas sample if the licensee was required to unload an MPC. The procedure addressed ALARA concerns that could develop when performing the activity.

Documents Reviewed: a) Callaway Procedure CSP-ZZ-07046, "MPC Boron and Gas Activity," Revision 0

Category: Unloading Operations **Topic:** Canister Reflooding

Reference: FSAR 1032, Section 9.4.3.5.c. Revision 3

Requirement Reflood the canister slowly with a pressure of less than 90 psi through the drain port until bubbling from the vent line has terminated.

Observation: Reflooding of a canister would be performed through the drain port at a pressure of less than 90 psig. Procedure HPP-2253-500, Step 7.10.12 described commencing the reflooding of the canister. A note above that step required monitoring the pressure gage to ensure the MPC remained less than 90 psig. Attachment 8.8 of the procedure

contained the MPC reflood arrangement diagram that showed water would enter the MPC via the drain port.

Documents Reviewed: a) Holtec Procedure, HPP-2253-500 "MPC Unloading at Callaway," Revision 7

Category: Unloading Operations **Topic:** Hydrogen Monitoring

Reference: FSAR 1032, Table 9.1.1 Revision 3

Requirement To preclude the potential for hydrogen ignition during lid cutting, operating procedures require monitoring for combustible gas and purging the space beneath the canister lid with an inert gas.

Observation: Hydrogen monitoring was required during lid cutting operations and was incorporated into the weld cutting dry run that took place at Holtec HMD facility on June 16-18, 2015. NRC inspectors observed the weld cutting dry run and use of the hydrogen monitoring system. Holtec procedure HPP-2253-500 required continuous monitoring for hydrogen during cutting operations in Step 7.13.21. Personnel were required to record the hydrogen levels in Attachment 8.14, "MPC Combustible Gas Monitoring and Argon Purge Log," every 20 minutes during the duration of the cutting evolution until cutting was complete.

Documents Reviewed: a) Holtec Procedure HPP-2253-500, "MPC Unloading at Callaway," Revision 7

Category: Welding **Topic:** Closure Ring, Vent and Drain Port Plate Weld PT

Reference: CoC 1040, Appendix B, Table 3-1 Amendment 0

Requirement A liquid penetrant (PT) examination is required on the root (if more than one weld pass is required) and the final pass on the vent and drain port cover plate welds. The PT examination shall be performed in accordance with NB-5245.

Observation: The PCI project instruction, in the steps under section 9.0, established a process for examining and documenting the acceptability of all levels of welding taking place to secure the closure lid and port vent cap lids to the Holtec Multi-Purpose Canister. Those steps of the instruction direct a liquid dye penetrant test after the root and other closure passes, after hydro testing, and prior to helium leak testing for the drain port cover plates. The NDE examinations for all sections of the final closure welds were spelled out by procedure and instructions. The PCI procedure satisfied the criteria set forth in CoC 1040 regarding PT of the closure lid, rings, and vent port caps.

Documents Reviewed: a) PCI Project Instruction PI-CNSTR-OP-CAL-H-01, "Closure Welding of Holtec Multi-Purpose Canisters – UMAX," Revision 1

Category: Welding **Topic:** Combustible Gas Monitoring

Reference: CoC 1040, Appendix B, Section 3.5 Amendment 0

Requirement During canister lid-to-shell welding operations, combustible gas monitoring of the space under the lid is required to ensure that there is no combustible gas mixture present in the welding area.

Observation: The requirement to monitor the space under the lid for combustible gases (hydrogen)

was covered in the PCI procedure for the closure welding of the MPC in Step 8.3.4 which calls for hydrogen monitoring the be performed until the lid-to-shell weld has been completed. PCI had procedures in place that met the combustible gas monitoring requirement as called out in the Holtec CoC 1040.

Documents Reviewed: a) PCI Project Instruction, PI-CNSTR-OP-CAL-H-01, "Closure Welding of Holtec Multi-Purpose Canisters – UMAX," Revision 1

Category: Welding **Topic:** Control of Weld Filler Materials

Reference: 10 CFR 72.154 Published 2015

Requirement The licensee shall establish measures to ensure that purchased material, equipment, and services conform to procurement documents. These measures must include provisions for source evaluation and selection, objective evidence of quality furnished by the contractor/subcontractor, inspection at the contractor/subcontractor source and examination of product on delivery. Records shall be available for the life of the ISFSI. The effectiveness of the control of quality by contractors/subcontractors shall be assessed at intervals consistent with the importance, complexity and quantity of the product or service.

Observation: PCI had established for Callaway a procedure for control of weld wire and other materials that met the applicable requirements of 10 CFR 72.154. PCI Procedure GQP-7.1 covered the procurement, receipt, storage, and issuance requirements for weld and fill materials to be used in support of dry fuel storage operations at Callaway. WCP-3 implemented the security and storage requirements of NUREG/CR-6314 Section 4.3.2.1.2 (6) (d) in its Section 8, "Storage and Disbursement Areas." This section of the PCI procedure specified how welding filler materials, electrodes, and other materials shall be stored and accessed by PCI welders and others who needed to access these materials.

Documents Reviewed: a) PCI Procedure GQP-7.1, "Procurement, Receipt, Storage, and Issue of ASME III, Subsection NCA 3800 Weld Materials," Revision 7; b) PCI Welding Control Procedure, WCP-3, "Weld Material Control," Revision 0

Category: Welding **Topic:** Minimum Delta Ferrite Content

Reference: ASME Section III, Article NB-2433; Reg Guide 1.31 Published 2007

Requirement A delta ferrite determination must be made for A-No.8 consumable inserts, bare electrode, rod, or wire filler metal. Exceptions: 1) A-No.8 metal used for weld metal cladding; 2) SFA-5.4 and SFA-5.9 metal; 3) Type 16-8-2 metal. The minimum acceptable delta ferrite content is 5 FN and it must be stated in the certification records.

Observation: The ferrite number for all of the inspected certified mill test results (CMTRs) were above the required number of 5 FN. The weld filler material's CMTRs inspected by the NRC all exceeded the minimum code requirements.

Documents Reviewed: a) PCI Certificate of Conformance 907864-01; b) ARCOS Certified Material Test Report for PCI purchase order, PO# 4500651885

Category: Welding **Topic:** Procedure Qualification Record (PQR)

Reference: ASME Section IX, Part QW-200.2 Published 2007

Requirement Each manufacturer or contractor shall prepare a Procedure Qualification Record (PQR) for each procedure. The completed PQR shall document all essential and, when required, all supplementary essential variables of QW-250 through QW-280 for each welding process used during the welding of the test coupon. Non essential variables may be documented at the contractor's option. The PQR shall be certified accurate by the manufacturer or contractor.

Observation: PCI, the welding contractor for Callaway, had prepared the Procedure Qualification Records with the required information for each welding procedure. The procedure qualification records (PQR-062, PQR-063, PQR-600, PQR-864, and PQR-899) for Weld Procedure Specifications (WPS) 8-MC-GTAW and 8 MN-GTAW listed the proper essential, supplementary, and non-essential variables, as specified in ASME Section IX, Part QW-200.2

Documents Reviewed: a) PCI PQR-062, Revision 3; b) PCI PQR-600, Revision 6; c) PCI PQR-864, Revision 2; d) PCI PQR-899, Revision 0; e) PCI PQR-063, Revision 6; f) PCI WPS 8-MC-GTAW, Revision 15; g) PCI WPS 8 MN-GTAW, Revision 3

Category: Welding **Topic:** Procedure Qualification Tests

Reference: ASME Section III, Article NB-4331 Published 2007

Requirement All welding procedure qualification tests shall be in accordance with the requirements of Section IX. ASME Section IX Article II QW-202.2 (b) requires partial penetration groove welds to be qualified in accordance with the requirements of QW-451. QW-202.2 (c) states that welding procedure specification (WPS) qualification for fillet welds may be made on groove-weld test coupons using test specimens specified in (b) above.

Observation: The procedure qualification records (PQR-062, PQR-063, PQR-600, PQR-864, and PQR-899) test coupons, which qualified Welding Procedure Specifications 8-MC-GTAW and 8-MN-GTAW, all satisfactorily passed the required tests per Table QW-451.1 "Groove - Weld Tension Tests and Transverse-Bend Tests."

Documents Reviewed: a) PCI PQR-062, Revision 3; b) PCI PQR-600, Revision 6; c) PCI PQR-864, Revision 2; d) PCI PQR-899, Revision 0; e) PCI PQR-063, Revision 6; f) PCI WPS 8-MC-GTAW, Revision 15; g) PCI WPS 8 MN-GTAW, Revision 3

Category: Welding **Topic:** Tack Welds

Reference: ASME Section III, Article NB-4231.1 Published 2007

Requirement Tack welds used to secure alignment shall either be removed completely when they have served their purpose, or their stopping and starting ends shall be properly prepared by grinding or other suitable means so that they may be satisfactorily incorporated into the final weld. When tack welds are to become part of the finished weld, they shall be visually examined and defective tack welds shall be removed.

Observation: NRC reviewed PCI's general welding procedure and found that it did not explicitly address the incorporation of tack welds into the final weld by grinding or other suitable means. This was taken and placed into the Callaway corrective action program as CAR

201501626. The licensee decided that language would be adopted and placed into the general welding standard to address this shortcoming. Language was added to Step 7.2.5 in Revision 1 of the PCI procedure for closure welding of Holtec MPCs, directing that tack weld starts and stops should be "ground" or "feathered" for suitable incorporation into the final weld.

Documents Reviewed: a) PCI General Welding Standard - 1 (GWS-1), Revision 0; b) PCI Project Instruction PI-CNSTR-OP-CAL-H-01, "Closure Welding of Holtec Multi-Purpose Canisters – UMAX," Revision 1

Category: Welding **Topic:** Weld Repairs - Base Metal Defects

Reference: ASME Section III, Article NB-4132 Published 2007

Requirement Weld repairs exceeding in depth the lesser of 3/8 inch (10 mm) or 10 percent of the section thickness, shall be documented on a report which shall include a chart which shows the location and size of the prepared cavity, the welding material identification, the welding procedure, the heat treatment, and the examination results of the weld repair.

Observation: This requirement was met. PIC Weld Control Procedure, WCP-5, did not reference ASME Section III, Article NB-4132 as a standard, nor did it include the requirements that all weld repairs exceeding 3/8 inch or 10 percent of the section thickness shall be documented on a report that includes a chart showing the location and size of the prepared cavity, weld material identification, procedure and examinations results of the weld repair. This issue was addressed in a revision to PCI Project Instruction PI-CNSTR-OP-CAL-H-01. Revision 1 of the Procedure PI-CNSTR-OP-CAL-H-01 in Step 7.2.6 included wording to align the weld repair criteria at Callaway with the ASME code document. In addition, the base metal and weld repair worksheet, Attachment 10, of that instruction was updated, as well.

Documents Reviewed: a) PCI Project Instruction PI-CNSTR-OP-CAL-H-01, "Closure Welding of Holtec Multi-Purpose Canisters – UMAX," Revision 1; b) PCI Weld Control Procedure WCP-5, "Weld and Base Metal Repair," Revision 0

Category: Welding **Topic:** Weld Types for Canister Lid

Reference: FSAR 1032, Table 7.1.1 Revision 3

Requirement The canister closure welds on the canister lid shall be the following types: (a) canister lid to shell - partial penetration groove (b) vent and drain port cover plates - partial penetration groove (c) closure ring to shell - fillet (d) closure ring to closure ring radial - partial penetration groove (e) closure ring to lid - partial penetration groove.

Observation: PCI had procedures in place that insured all applicable welds listed in the Holtec HI-STORM FW FSAR Table 7.1.1 were accounted for. The types of canister closures welds used for the MPC were documented per the project instruction in Attachment 6, "PCI Energy Services Weld Process Traveler," and Attachment 7, "PCI Energy Services Multiple Weld Data Card." These attachments listed the weld types for each weld made in the process of final closure of the MPC.

Documents Reviewed: a) PCI Project Instruction PI-CNSTR-OP-CAL-H-01, "Closure Welding of Holtec Multi-Purpose Canisters – UMAX," Revision 1

Category: Welding **Topic:** Welder Performance Qualification Test

Reference: ASME Section IX, Part QW-301.2 Published 2007

Requirement The welder performance qualification test shall be welded in accordance with a qualified welding procedure specification (WPS), unless preheat or post weld heat treatment is specified.

Observation: NRC inspectors reviewed the Callaway procedure that provided instructions for qualifying welders, brazers, and brazing operators working under the Callaway's QA program. NRC inspectors also reviewed the welder performance qualification records (WPQs) and welding procedure specifications for the various welding procedures qualified for use during dry cask loading operations at Callaway. Each of the welders present during the welding dry run were qualified to perform the lid to shell, port caps covers, and final closure ring welding for the Holtec MPC. In addition, the qualifications of an additional welder who participated in the first loading campaign was also reviewed by the NRC inspectors.

Documents Reviewed: a) Callaway Procedure MTW-ZZ-WP002, "Welder Performance Qualification," Revision 27 b) PCI Welder Maintenance Logs (numerous); c) PCI ASME Section IX Welding Procedure Specifications for 8 MC-GTAW, Revision 15 and 8 MN-GTAW, Revision 4; d) PCI ASME IX Welding Procedure Qualification Records (numerous)

Category: Welding **Topic:** Welding Operator Performance Qualification

Reference: ASME Section IX, Parts QW-301.4, 361.2, 452.1, 6 Published 2007

Requirement The record of welding operator performance qualification (WOPQ) tests shall include the essential variables listed in QW-360, the type of test and test results, and the ranges qualified in accordance with QW-452. The essential variables for machine welding are: (1) welding process; (2) direct or remote visual control; (3) automatic arc voltage control (GTAW); (4) automatic joint tracking; (5) position qualified; (6) consumable inserts; (7) backing; and (8) single or multiple passes per side. Two side bend tests are required for groove weld test coupons 3/4 inch thick or greater. Groove weld tests qualify fillet welds.

Observation: NRC reviewed the welder performance qualification reports (WPQs) of several welders qualified to support dry cask storage operations at Callaway, including the welders who participated in the welding dry-run activities the week of May 18, 2015. All of the qualification records for welders explicitly addressed all eight essential variables noted in the requirement section, above. The welders performing the welding dry-run activities at Callaway were qualified in the eight essential variables. NRC inspectors also reviewed the performance qualification of an additional welder who participated in the initial loading campaign at Callaway who was not present during the welding dry-run.

Documents Reviewed: a) Numerous PCI Welder Performance Qualification records (WPQs)

Category: Welding **Topic:** Welding Procedure Specification (WPS)

Reference: ASME Section IX, Part QW-200.1 Published 2007

Requirement Each manufacturer or contractor shall prepare written Welding Procedure Specifications for making production welds to code requirements. Welding Procedure Specifications shall include the essential, non-essential, and (when required) supplementary essential variables for each welding process. The variables are listed in QW-250 through QW-280 and are defined in Article IV, Welding Data.

Observation: PCI, the welding contractor for Callaway, had prepared written Weld Procedure Specifications (WPS) 8-MC-GTAW and 8 MN-GTAW. Each WPS listed the proper essential, supplementary, and non-essential variables.

Documents Reviewed: a) PCI WPS 8-MC-GTAW, Revision 15; b) PCI WPS 8 MN-GTAW, Revision 3

CERTIFICATE OF SERVICE

I hereby certify that on July 20, 2020, I electronically filed the foregoing with the Clerk of the Court for the United States Court of Appeals for the Ninth Circuit by using the appellate CM/ECF system.

Participants in the case who are registered CM/ECF users will be served by the appellate CM/ECF system.

Dated: July 20, 2020

Respectfully submitted,
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No. 20-70899

IN THE UNITED STATES COURT OF APPEALS
FOR THE NINTH CIRCUIT

IN RE PUBLIC WATCHDOGS,

Petitioner,

v.

UNITED STATES NUCLEAR REGULATORY COMMISSION,

Respondent.

SOUTHERN CALIFORNIA EDISON COMPANY,
Intervenor.

INTERVENOR SOUTHERN CALIFORNIA EDISON'S SUPPLEMENTAL EXCERPTS OF RECORD

VOLUME 5 OF 8

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SUPPLEMENTAL EXCERPTS OF RECORD:

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FINAL SAFETY ANALYSIS REPORT

on

THE HI-STORM UMAX

CANISTER STORAGE SYSTEM

by

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3.	CHAP3	JZ	6/29/2012	75144	CB	11/20/12	165716	CB	7/11/14	500952
4.	CHAP4	YL	6/29/2012	187264	AHM	11/20/12	73565	AHM	7/11/14	432189
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5.	CHAP5	DM	6/29/2016	374676						
6.	CHAP6	N/A	N/A	N/A						
7.	CHAP7	N/A	N/A	N/A						
8.	CHAP8	KSM	6/29/2016	582053						
9.	CHAP9	KSM	6/29/2016	792428						
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11.	CHAP11	N/A	N/A	N/A						
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3. Revisions to this document may be made by adding supplements to the document and replacing the "Table of Contents", this page and the "Revision Log".

GLOSSARY OF TERMS USED IN HI-STORM SAFETY ANALYSIS REPORTS

AFR is an acronym for Away from Reactor storage**.**

ALARA is an acronym for As Low As Reasonably Achievable

Ambient Temperature for Short Term Operations (operations involving use of the HI-TRAC, a Lifting device and/ or a on-site transport device) is defined as the 24 hour average of the local temperature as forecast by the National Weather Service.

Ancillary or Ancillary Equipment is the generic name of a device used to carry out "short term operations.

Bottom Lid means the removable lid that fastens to the bottom of the HI-TRAC transfer cask body to create a gasketed barrier against in-leakage of pool water in the space around the MPC.

Bottom MPC Guides are the MPC Guides located in the bottom region of the MPC storage cavity.

BWR is an acronym for Boiling Water Reactor.

Canister means an all-welded vessel containing used fuel that has been qualified to serve as a confinement boundary under the rules of 10CFR 72.

Cavity Enclosure Container (CEC) means a thick walled cylindrical steel weldment that defines the storage cavity for the MPCs.

CG is an acronym for center of gravity.

Closure Lid means the METCON lid that is installed on the MPC storage cavity to provide physical and shielding protection to the stored MPC.

Commercial Spent Fuel or CSF refers to nuclear fuel used to produce energy in a commercial nuclear power plant.

Confinement Boundary means the outline formed by the sealed, cylindrical enclosure of the Multi-Purpose Canister (MPC) shell welded to a solid baseplate, a lid welded around the top circumference of the shell wall, the port cover plates welded to the lid, and the closure ring welded to the lid and MPC shell providing the redundant sealing.

Confinement System means the Multi-Purpose Canister (MPC) which encloses and confines the spent nuclear fuel during storage.

Container Flange means the ring flange that is welded to the upper extremity of the Container shell.

Container Shell means the cylindrical portion of the Cavity Enclosure Container

Controlled Area means that area immediately surrounding an ISFSI for which the owner/user exercises authority over its use and within which operations are performed.

Controlled Low-Strength Material (CLSM) Withheld in Accordance with 10 CFR 2.390

Cooling Time (or post-irradiation cooling time) for a spent fuel assembly is the time between its final discharge from the reactor to the time it is loaded into the MPC.

Critical Characteristic means a feature of a component or assembly that is necessary for the proper safety function of the component or assembly. Critical characteristics of a material are those attributes that have been identified, in the associated material specification, as necessary to render the material's intended function.

DAS is the abbreviation for the Decontamination and Assembly Station. It means the location where the Transfer Cask is decontaminated and the MPC is processed (i.e., where all operations culminating in lid and closure ring welding are completed).

DBE means Design Basis Earthquake.

DCSS is an acronym for Dry Cask Storage System.

Damaged Fuel Assembly is a fuel assembly with known or suspected cladding defects, as determined by review of records, greater than pinhole leaks or hairline cracks, empty fuel rod locations that are not replaced with dummy fuel rods, missing structural components such as grid spacers, whose structural integrity has been impaired such that geometric rearrangement of fuel or gross failure of the cladding is expected based on engineering evaluations, or those that cannot be handled by normal means. Fuel assemblies that cannot be handled by normal means due to fuel cladding damage are considered fuel debris.

Damaged Fuel Container (or Canister) or DFC means a specially designed enclosure for damaged fuel or fuel debris which permits flow of gaseous and liquid media while minimizing dispersal of gross particulates.

Design Basis Load (DBL) is a loading defined in this SAR to bound one or more events that are applicable to the storage system during its service life. Thus, the pressure loading on the cask's lid specified in this SAR is a DBL because it is set substantially above the pressure from accumulated snow set down in the national consensus standard for the 48 contiguous United States.

Design Heat Load or Design Basis Heat Load is the computed heat rejection capacity of the HI-STORM system with a certified MPC loaded with CSF stored in uniform storage with the ambient at the normal temperature and the peak cladding temperature (PCT) at 400ºC. The Design Heat Load is less than the thermal capacity of the system by a suitable margin that reflects the conservatism in the system thermal analysis.

Design Life is the minimum duration for which the component is engineered to perform its intended function set forth in this SAR, if operated and maintained in accordance with this SAR.

Design Report is a document prepared, reviewed and QA validated in accordance with the provisions of 10CFR72 Subpart G. The Design Report shall demonstrate compliance with the requirements set forth in the Design Specification. A Design Report is mandatory for systems, structures, and components designated as Important to Safety. The SAR serves as the Design Report for the HI-STORM UMAX System.

Design Specification is a document prepared in accordance with the quality assurance requirements of 10CFR72 Subpart G to provide a complete set of design criteria and functional requirements for a system, structure, or component, designated as Important to Safety, intended to be used in the operation, implementation, or decommissioning of the HI-STORM UMAX System. The SAR serves as the Design Specification for the HI-STORM UMAX System.

Divider Shell means a cylindrical shell bearing insulation over most of its inner or outer surface that divides the annular space between the MPC and the CEC shell into two discrete regions for down- flow and up-flow of air..

Duct Extension means a removable, non-structural member fastened to the inlet or outlet duct to move the location of air intake or exhaust, as applicable

Enclosure Wall means an optional circumscribing structure installed to mitigate the incursion of groundwater into the subgrade space of the ISFSI.

Enclosure Vessel (or MPC Enclosure Vessel) means the pressure vessel defined by the cylindrical shell, baseplate, port cover plates, lid, closure ring, and associated welds that provides confinement for the helium gas contained within the MPC. The Enclosure Vessel (EV) and the fuel basket together constitute the multi-purpose canister.

Equivalent (or Equal) Material is a material with critical characteristics (see definition above) that meet or exceed those specified for the designated material.

Fracture Toughness is a property which is a measure of the ability of a material to limit crack propagation under a suddenly applied load.

FSAR is an acronym for Final Safety Analysis Report (10CFR72).

Fuel Basket means a honeycombed structural weldment with square openings which can accept a fuel assembly of the type for which it is designed.

Fuel Building is the generic term used to denote the building in which the fuel loading and where a portion of "short-term operations" will occur.

Fuel Debris is ruptured fuel rods, severed rods, loose fuel pellets, containers or structures that are supporting these loose fuel assembly parts, or fuel assemblies with known or suspected defects which cannot be handled by normal means due to fuel cladding damage.

Fuel Shim is a suitably sized metallic part interposed in the space between the fuel and the MPC cavity at either the top or the bottom (or both) ends of the fuel to minimize the axial displacement of the SNF within the MPC due to longitudinal inertia forces.

High Burnup Fuel, or HBF is a commercial spent fuel assembly with an average burn-up greater than 45,000 MWD/MTU.

HI-PORT is a Holtec trade name for an engineered Low Profile Transporter to that maximizes protection of the cask against overturning under seismic conditions.

HI-TRAC is a generic term for the transfer cask used to house the MPC during MPC fuel loading, unloading, drying, sealing, and on-site transfer operations to a HI-STORM storage module or HI-STAR storage/transportation overpack. The HI-TRAC shields and protects the loaded MPC during short term operations.

HI-STORM VVM means the module that receives and contains the sealed multi-purpose canisters containing spent nuclear fuel for long term storage. It provides the gamma and neutron shielding, ventilation passages, missile protection, and protection against natural phenomena and accidents for the loaded MPC.

HI-STORM UMAX system consists of loaded MPCs stored in the HI-STORM UMAX VVM .

HI-STORM 100 System consists of any loaded MPC model placed within any design variant of the HI-STORM overpack in Docket number 72-1014.

Holtite*™*-A is a trademarked Holtec International neutron shield material.

Important to Safety (ITS) means a function or condition required to store spent nuclear fuel safely; to prevent damage to spent nuclear fuel during handling and storage, and to provide reasonable assurance that spent nuclear fuel can be received, handled, packaged, stored, and retrieved without undue risk to the health and safety of the public.

Independent Spent Fuel Storage Installation (ISFSI) means a facility designed, constructed, and licensed for the interim storage of spent nuclear fuel and other radioactive materials associated with spent fuel storage in accordance with 10CFR72.

Intact Fuel Assembly is defined as a fuel assembly without known or suspected cladding defects greater than pinhole leaks and hairline cracks, and which can be handled by normal

means. Fuel assemblies without fuel rods in fuel rod locations shall not be classified as Intact Fuel Assemblies unless dummy fuel rods are used to displace an amount of water greater than or equal to that displaced by the fuel rod(s).

Interim Storage means an autonomous monitored canister storage facility from which the stored canister can be retrieved, if necessary.

Interfacing Components means the weldments certified in other dockets that will be used with the HI-STORM UMAX VVM assemblies for transferring and storing MPCs in HI-STORM UMAX. The MPC is an Interfacing Component

ISFSI Pad means the reinforced concrete pad that provides the support surface for the cask handling device.

License Life means the duration for which the system is authorized by virtue of its certification by the U.S. NRC.

Licensing Drawings or Licensing Drawing Package is an integral part of this SAR wherein the essential geometric and material information on HI-STORM UMAX is compiled to enable the safety evaluations pursuant to 10 CFR 72 to be carried out.

Long-term Storage means the time beginning after on-site handling is complete and the loaded overpack is at rest in its designated storage location on the ISFSI pad and lasting up to the end of the licensed life of the HI-STORM 100 System.

Low Profile Transporter (LPT) is the generic name of the ancillary used to move a loaded or empty cask in a plant's "truck bay" and/or the haul path with the cask directly situated on a low lying platform founded on a structurally robust frame such that an uncontrolled lowering (free fall) of the cask is not credible. The LPT must be sufficiently short to insure that the loaded cask can clear the roll-up door in the truck bay.

Lowest Service Temperature (LST) is the minimum metal temperature of a part for the specified service condition.

Maximum Reactivity means the highest possible k-effective including bias, uncertainties, and calculational statistics evaluated for the worst-case combination of fuel basket manufacturing tolerances.

METAMIC® is a trade name for an aluminum/boron carbide composite neutron absorber material qualified for use in the MPCs and in wet storage applications.

METAMIC-HT is the tradename for the metal matrix composite made by imbedding nano-particles of aluminum oxide and fine boron carbide powder on the grain boundaries of aluminum resulting in improved structural strength properties at elevated temperatures.

METCON□ is a trade name for the HI-STORM overpack. The trademark is derived from the

metal-**con**crete composition of the HI-STORM overpack.

MGDS is an acronym for Mined Geological Disposal System.

Minimum Enrichment is the minimum assembly average enrichment. Natural uranium blankets are not considered in determining minimum enrichment.

Moderate Burnup Fuel, or MBF is a commercial spent fuel assembly with an average burnup less than or equal to 45,000 MWD/MTU.

Multi-Purpose Canister or MPC means the sealed canister consisting of a fuel basket for spent nuclear fuel storage, contained in a cylindrical canister shell (the MPC Enclosure Vessel). The MPC is the confinement boundary for storage conditions.

MPC Guides is a generic term to represent Top or Bottom MPC Guides

MPC Transfer means transfer of the MPC between the storage module and the transfer cask which begins when the MPC is lifted off the HI-TRAC bottom lid and ends when the MPC is supported from beneath by the module (or the reverse).

NDT is an acronym for Nil Ductility Transition Temperature, which is defined as the temperature at which the fracture stress in a material with a small flaw is equal to the yield stress in the same material if it had no flaws.

Neutron Absorber is a generic term used in this SAR to indicate any neutron absorber material qualified for use in the MPCs certified in a HI-STORM docket.

Neutron Shielding means a material used to thermalize and capture neutrons emanating from the radioactive spent nuclear fuel.

Non-Fuel Hardware is defined as Burnable Poison Rod Assemblies (BPRAs), Thimble Plug Devices (TPDs), Control Rod Assemblies (CRAs), Axial Power Shaping Rods (APSRs), Wet Annular Burnable Absorbers (WABAs), Rod Cluster Control Assemblies (RCCAs), Control Element Assemblies (CEAs), Neutron Source Assemblies (NSAs), water displacement guide tube plugs, orifice rod assemblies, Instrument Tube Tie Rods (ITTRs), vibration suppressor inserts, and components of these devices such as individual rods.

Planar-Average Initial Enrichment is the average of the distributed fuel rod initial enrichments within a given axial plane of the assembly lattice.

Plain Concrete is concrete that is unreinforced and is of density specified in this FSAR.

Post-Core Decay Time (PCDT) is synonymous with cooling time.

PWR is an acronym for pressurized water reactor.

Reactivity is used synonymously with effective neutron multiplication factor or k-effective.

Regionalized Fuel Loading is a term used to describe an optional fuel loading strategy used in lieu of uniform fuel loading wherein the storage locations are ascribed to two or more distinct regions each with its own maximum allowable specific heat generation rate.

Regionalized Fuel Storage is a term used to describe an optimized fuel loading strategy wherein the storage locations are ascribed to distinct regions each with its own maximum allowable specific heat generation rate.

Removable Shielding Girdle is an ancillary designed to be installed to provide added shielding to the personnel working in the top region of the transfer cask.

SAR is an acronym for Safety Analysis Report.

Self-hardening Engineered Subgrade (or SES) means CLSM or lean concrete in this FSAR.

Service Life means the duration for which the component is reasonably expected to perform its intended function, if operated and maintained in accordance with the provisions of this FSAR. Service Life may be much longer than the Design Life because of the conservatism inherent in the codes, standards, and procedures used to design, fabricate, operate, and maintain the component.

Short-term Operations means those normal operational evolutions necessary to support fuel loading or fuel unloading operations. These include, but are not limited to MPC cavity drying, helium backfill, MPC transfer, and onsite handling of a loaded HI-TRAC transfer cask.

Single Failure Proof means that the handling system is designed so that all directly loaded tension and compression members are engineered to satisfy the enhanced safety criteria of Paragraphs 5.1.6(1)(a) and (b) of NUREG-0612.

SNF is an acronym for spent nuclear fuel.

Specific Heat Load means the heat emission rate from one fuel assembly. Sum of the Specific Heat loads of all fuel assemblies in a canister is referred to as the Aggregate heat Load or SFSC Heat Load.

SSC is an acronym for Structures, Systems and Components.

STP is Standard Temperature and Pressure conditions.

Support Foundation Pad (SFP) means the reinforced concrete pad located underground on which the CECs are situated.

Subgrade is the lateral space between each CEC, the SFP and the ISFSI Pad.

TAL is an acronym for the Tapped Anchor Location.

Thermal Capacity of the HI-STORM system is defined as the amount of heat the storage system, containing an MPC loaded with CSF stored in *uniform storage,* will actually reject with the ambient environment at the normal temperature and the peak fuel cladding temperature (PCT) at 400ºC.

Thermo-siphon is the term used to describe the buoyancy-driven natural convection circulation of helium within the MPC fuel basket.

Top MPC Guides and Bottom MPC Guides mean the set of radial plates that are shaped to aid in the insertion and withdrawal of MPCs and serve to restrain the MPC's lateral movement during seismic events.

TOG is an acronym for top-of-the-grade of the ISFSI and identified by the by the riding surface of the cask transporter.

Traveler means the set of sequential instructions used in a controlled manufacturing program to ensure that all required tests and examinations required upon the completion of each significant manufacturing activity are performed and documented for archival reference.

Undamaged Fuel Assembly is defined as a fuel assembly without known or suspected cladding defects greater than pinhole leaks and hairline cracks, and which can be handled by normal means. Fuel assemblies without fuel rods in fuel rod locations shall not be classified as Intact Fuel Assemblies unless dummy fuel rods are used to displace an amount of water greater than or equal to that displaced by the fuel rod(s).

Under-grade is the space below the SFP**.**

Uniform Fuel Loading is a fuel loading strategy where any authorized fuel assembly may be stored in any fuel storage location, subject to other restrictions in the CoC, such as those applicable to non-fuel hardware, and damaged fuel containers.

Vertical Cask Transporter or VCT is the generic name for a device that has the ability to raise or lower a cask or a canister with the built-in safety of a redundant drop protection system. A VCT may be designed to be limited in its operation space to the ISFSI pad area and/or it may have the capability to translocate the cask over a suitably engineered haul path.

VVM is an acronym for Vertical Ventilated Module

ZPA is an acronym for zero period acceleration.

ZR means any zirconium-based fuel cladding material authorized for use in a commercial nuclear power plant reactor. Any reference to Zircaloy fuel cladding in this FSAR applies to any zirconium-based fuel cladding material.

FSAR SECTION REVISION STATUS, LIST OF AFFECTED SECTIONS AND REVISION SUMMARY

FSAR Report No.: HI-2115090 | **FSAR Revision Number:** 3

FSAR Title: Final Safety Analysis Report on the HI-STORM UMAX System

This FSAR is submitted to the USNRC in support of Holtec International's application to secure a CoC under 10CFR Part 72.

FSAR review and verification are controlled at the chapter level and changes are annotated at the chapter level.

A section in a chapter is identified by two numerals separated by a decimal. Each section begins on a fresh page. Unless indicated as a "complete revision" in the summary description of change below, if any change in the content is made, then the change is indicated by a "bar" in the right page margin and the revision number of the entire chapter including applicable figures (annotated in the footer) is changed.

A summary description of change is provided below for each FSAR chapter. Minor editorial changes to this FSAR may not be summarized in the description of change.

Chapter 1

Affected Section or Table No.	Current Revision No.	Summary Description of Change
Section 1.0	3	Updated description of underground per ECO
Section 1.0.2		Updated per Amendment 1 changes
Section 1.1		Updated per ECO on shell thickness
Section 1.2.2		Updated VVM description per ECO
Section 1.5		Drawings updated per ECOs

Chapter 2

Section or Table No.	Current Revision No.	Summary Description of Change
Section 2.0.2	3	Updated SSI discussion per ECO
Table 2.0.5		Updated ITS category per ECO
Section 2.2		Updated per Amendment 1 changes
Section 2.3.3.3		Updated discussion per ECO

Table 2.3.1		Clarified licensed life per ECO
Table 2.3.2		Properties updated per ECO
Table 2.3.7		Updated design temperatures per ECO
Table 2.3.10		Updated per Amendment 1 changes
Section 2.4.3		Updated per Amendment 1 changes
Table 2.4.5		Updated per Amendment 1 changes
Figure 2.4.4		Notes updated per ECO
Figures 2.4.5 through 2.4.11		Updated per Amendment 1 changes
Section 2.13		Updated per Amendment 1 changes
Chapter 3		
Section or Table No.	**Current Revision No.**	**Summary Description of Change**
Table 3.1.13	3	Updated properties and notes per ECOs
Table 3.1.15		Updated notes per ECO
Section 3.4.4.4		Updated per Amendment 1 changes
Section 3.4.7		Updated licensed life and materials per ECOs
Tables 3.4.10 through 3.4.13		Updated per Amendment 1 changes
Figures 3.4.23 through 3.4.29		Updated per Amendment 1 changes
Section 3.8		Updated per Amendment 1 changes
Chapter 4		
Section or Table No.	**Current Revision No.**	**Summary Description of Change**
Table 4.1.1	3	Shell thickness updated per ECO
Table 4.2.5		Notes updated per ECO
Chapter 5		
Section or Table No.	**Current Revision No.**	**Summary Description of Change**

Section 5.0 Section 5.3 Table 5.3.2 Table 5.3.3	3	Editorial change per ECO Density discussion updated per ECO Notes updated per ECO Materials updated per ECO
Chapter 6		
Section or Table No.	**Current Revision No.**	**Summary Description of Change**
	2	No changes
Chapter 7 Changes		
Section or Table No.	**Current Revision No.**	**Summary Description of Change**
	2	No changes
Chapter 8 Changes		
Section or Table No.	**Current Revision No.**	**Summary Description of Change**
Section 8.1 Table 8.1.3 Section 8.2.1 Section 8.7 Section 8.11	3	FW FSAR References updated per ECO Materials updated per ECO Materials updated per ECO Materials updated per ECO Materials updated per ECO
Chapter 9 Changes		
Section or Table No.	**Current Revision No.**	**Summary Description of Change**
Section 9.2.1 Section 9.2.3 Section 9.4.2	3	VCT movement description updated per ECO MPC plugs updated per ECO MPC plugs updated per ECO
Chapter 10 Changes		
Section or Table No.	**Current Revision No.**	**Summary Description of Change**

Table 10.1.1 Section 10.2	3	Fit-up description updated per ECO Editorial change per ECO
Chapter 11 Changes		
Section or Table No.	**Current Revision No.**	**Summary Description of Change**
	2	No changes
Chapter 12 Changes		
Section or Table No.	**Current Revision No.**	**Summary Description of Change**
	2	No changes
Chapter 13 Changes		
Section or Table No.	**Current Revision No.**	**Summary Description of Change**
Section 13.A	3	Updated LCO basis per ECO
Chapter 14 Changes		
Section or Table No.	**Current Revision No.**	**Summary Description of Change**
	0	No Changes

HI-STORM UMAX FSAR
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CHAPTER 1: GENERAL DESCRIPTION OF THE HI-STORM UMAX SYSTEM

1.0 GENERAL INFORMATION

This final safety analysis report (FSAR) describes the Holtec International HI-STORM UMAX Canister Storage System (HI-STORM UMAX) and contains the necessary information and analyses to support a United States Nuclear Regulatory Commission (USNRC) licensing review as a spent nuclear fuel (SNF) dry storage cask under the provisions of 10 CFR 72 [1.0.3]. This report, prepared pursuant to 10 CFR 72.230, describes the basis for NRC approval and issuance of a Certificate of Compliance (CoC) on the HI-STORM UMAX System under 10 CFR 72, Subpart L to safely store spent nuclear fuel (SNF) at an Independent Spent Fuel Storage Installation (ISFSI) under the general license authorized by 10 CFR 72, Subpart K.

The HI-STORM UMAX stores a hermetically sealed canister containing spent nuclear fuel in an in-ground Vertical Ventilated Module (VVM). The safety evaluation and regulatory control is maintained in USNRC docket # 72-1040. The annex identifier UMAX is an acronym of Underground MAXimum capacity. HI-STORM UMAX is designed to provide long-term underground∗ storage of loaded multi-purpose canisters (MPC) previously certified for storage by the USNRC in Holtec International ("Holtec") Docket 72-1032 (HI-STORM FW) [1.0.2]. The HI-STORM UMAX VVM is essentially the underground equivalent of the HI-STORM FW overpack certified in Docket# 72-1032. Although the storage cavity dimensions and the air ventilation system in the HI-STORM UMAX VVM have been selected to enable it to also store all MPCs certified in for storage in the HI-STORM 100 overpack (Docket number 72-1014)[1.0.1], this FSAR does not seek to support their certification at this time. Safety analyses and evaluations of the HI-STORM 100 MPCs under (hypothetical) storage in HI-STORM UMAX are nevertheless included in this FSAR, as appropriate, to provide a comparative reference for the licensing-basis analyses of the HI-STORM FW canisters (MPC-37 & MPC-89). Thus while the safety analyses have been carried out in this FSAR for all MPCs presently certified in docket # 72-1014 (HI-STORM 100) and docket # 72-1032 (HI-STORM FW) , the Certificate-of-Compliance sought pursuant to this licensing submittal is *limited to qualifying only MPC-37 and MPC-89 which have been previously certified in the HI-STORM FW docket*. .Inclusion of the smaller MPCs originally certified in the HI-STORM 100 docket serves to underscore the plausible result that the structural and thermal margins are controlled by the larger canister. Specifically, it is found that MPC-37 governs in respect of structural and thermal margins. MPC-37 is therefore designated as the "governing canister". To insure that the CoC for the HI-STORM UMAX system is autonomously complete, the entire body of the latest versions

∗ The term "underground" is used throughout this document to refer to the HI-STORM UMAX System, in which MPCs are stored within a monolithic block, some of which is below grade, instead of within discrete above-ground overpacks. The term underground is not intended to mean that all portions of the system are located below the site grade level. Sites construct the HI-STORM UMAX in accordance with the license requirements and site-specific conditions.

of the Technical Specification pertaining to MPC-37, MPC-89 and HI-TRAC VW is excerpted from the "FW" docket and included in the proposed Technical Specifications for the HI-STORM UMAX storage system. To facilitate convenient access to the referenced material, the latest revision of the HI-STORM FW FSAR has been placed in this docket and a list of "FW " FSAR sections germane to this chapter is provided in a tabular form. Table 1.0.1 provides a listing of the applicable material adopted in this chapter by reference to the HI-STORM FW FSAR.

Except in cases where clarity of presentation calls for reproduction of the HI-STORM FW FSAR material in this FSAR, the safety analyses for the canisters and the transfer cask documented in the HI-STORM FW FSAR, where applicable, are incorporated by reference. *Directly copied FSAR text matter from the "FW" docket is provided in the "Arial" font to identify its provenance*.

HI-STORM UMAX is intended for dry storage of spent nuclear fuel at an Independent Spent Fuel Storage Installation (ISFSI) under 10CFR 72 [1.0.3] Subpart L. The HI-STORM UMAX ISFSI (illustrated in Figure 1.0.1), which may contain any number of UMAX VVMs, may be co-located at a licensed reactor site or at an away-from-reactor (AFR) site such as an interim storage facility. Certain innovative design features of HI-STORM UMAX are subject to patent action by the USPTO under application number 61625869 dated April 18, 2012. A licensing drawing package depicting the essential geometric details of the HI-STORM UMAX is provided in Section 1.5. The licensing drawings for MPC-37, MPC-89 and the HI-TRAC transfer cask are also reproduced in Section 1.5 from the "FW" docket for configuration control. The glossary preceding this chapter contains the definition of terms and acronyms that may be consulted as necessary.

This safety evaluation follows the guidelines of RG 3.61 [1.0.4] and NUREG-1536 [1.0.5] to qualify the MPC-37 and MPC-89 models previously certified in the HI-STORM FW Holtec docket for storage in the HI-STORM UMAX. This report has been prepared in the format and content suggested in NRC Regulatory Guide 3.61 [1.0.4] and NUREG-1536 Standard Review Plan for Dry Cask Storage Systems [1.0.5]. The only deviation in the format from the formatting instruction in Reg. Guide 3.61 is the insertion of a chapter (Chapter 8) on material compatibility pursuant to ISG-15 and renumbering of all subsequent chapters and only deviation from NUREG-1536 is the order of Chapters 5, 6 and 7 which is similar to HI-STORM FW format.

The Glossary contains a listing of the terminology and notation used in this FSAR.

The safety evaluations in this FSAR are intended to bound the conditions that exist in the vast majority of domestic power reactor sites and potential away-from-reactor storage sites in the contiguous United States. This includes the potential fuel assemblies which will be loaded into the system and the environmental conditions in which the system will be deployed. This FSAR also provides the basis for component fabrication and acceptance, and the requirements for safe operation and maintenance of the components, consistent with the design bases and safety analyses documented herein. In accordance with 10CFR72, Subpart K, site-specific implementation of the generically certified HI-STORM FW System requires that the licensee perform a site-specific evaluation, as defined in 10CFR72.212. The HI-STORM UMAX System FSAR identifies a number of conditions that are site-specific and are to be addressed in the licensee's 10CFR72.212 evaluation. These include:

- Siting of the ISFSI and design of the storage and security system. Site-specific demonstration of compliance with regulatory dose limits. Implementation of a site-specific ALARA program.
- An evaluation of site-specific hazards and design conditions that may exist at the ISFSI site or the transfer route between the plant's cask receiving bay and the ISFSI. These include, but are not limited to, explosion and fire hazards, flooding conditions, landslides, and lightning protection.
- Determination that the physical and nucleonic characteristics and the condition of the SNF assemblies to be stored meet the fuel acceptance requirements of the Certificate of Compliance.
- An evaluation of interface and design conditions that exist within the plant's Fuel Building in which canister fuel loading, canister closure, and canister transfer operations are to be conducted in accordance with the applicable 10CFR50 requirements and technical specifications for the plant.
- Detailed site-specific operating, maintenance, and inspection procedures prepared in accordance with the generic procedures and requirements provided in Chapters 9 and 10, and the Certificate of Compliance.
- Performance of pre-operational testing.
- Implementation of a safeguards and accountability program in accordance with 10CFR73. Preparation of a physical security plan in accordance with 10CFR73.55.
- Review of the reactor emergency plan, quality assurance (QA) program, training program, and radiation protection program.

In presenting the bounding generic analyses of this safety report, selected conditions are drawn from authoritative sources such as Regulatory Guides and NUREGs, where available.

Within this report, all figures, tables and references cited are identified by the double decimal system *m.n.i*, where *m* is the chapter number, *n* is the section number, and *i* is the table number. For a complete listing of Tables and Figure please consult the Table of Contents. For example, Figure 1.2.1 is the first figure in Section 1.2 of Chapter 1. Similarly, the following convention is used in the organization of chapters:

a. A chapter is identified by a whole numeral, say *m* (i.e., *m*=3 means Chapter 3)

b. A section is identified by one decimal separating two numerals. Thus, Section 3.1 is section 1 in Chapter 3.

c. A subsection has three numerals separated by two decimals. Thus, Subsection 3.2.1 is subsection 1 in Section 3.2.

d. A paragraph is denoted by four numerals separated by three decimals. Thus, Paragraph 3.2.1.1 is paragraph 1 in Subsection 3.2.1.

e. A subparagraph has five numerals separated by four decimals. Thus, Subparagraph 3.2.1.1.1 is subparagraph 1 in Paragraph 3.2.1.1.

Tables and figures associated with a section are placed after the text narrative. Drawings are controlled separately within the Holtec QA program and have individual revision numbers and are included in Section 1.5.

1.0.1 Engineering Change Orders

The changes authorized by the Holtec ECOs (with corresponding 10CFR72.48 evaluations, if applicable) listed in the following table are reflected in this Revision of the FSAR.

LIST OF ECO'S AND APPLICABLE 10CFR72.48 EVALUATIONS

Affected Item	ECO Number	72.48 Evaluation or Screening Number
MPC-37 Fuel Basket	**102-18**	**1110R0 and R1**
MPC-89 Fuel Basket	**101-17**	**1112**
	101-18R0 and R1	**1110R0 and R1**
MPC Enclosure Vessels (37 & 89)	**101-8R0 and R1; 101-9R0 and R1; 101-13; 102-8R0, R1, and R3; 102-10R0 and R1**	**1053R0 and R1**
	101-12; 102-11R0 and R1	**1076**
	102-13, 102-14	**1091**
	102-15	**1098**
	101-16; 102-17R0 and R1	**1103R0 and R1**
	101-19; 102-19	**1138**
	101-21; 102-21	**1164**
	101-15	**1093**
	101-17	**1112**
HI-TRAC VW – MPCs (37 & 89)	**103-11**	**1064**
	103-12	**1098**
	103-14	**1114**
HI-STORM UMAX Canister Storage System	**105-8**	**1083**
	105-9 R0, R1 and, R2	**1098**
	105-7	**N/A**

	105-10 R0 and R1	**N/A**
	105-13	**N/A**
General FSAR Changes	**5021-10 R0 and R1**	**1079**
	5021-11	**1098**
	5021-12	**1100**
	5021-14	**N/A**
	5021-15	**N/A**
	5021-16	**1125**
	5021-17	**1135**
	5021-18	**1139**
	5021-19	**N/A**
	5021-20	**1154**
	5021-21	**1174**
	5021-22	**1195**

1.0.2 Changes Introduced in this FSAR

This FSAR revision corresponds to Amendment #1 to the CoC of HI-STORM UMAX Canister Storage system. The changes in this FSAR are necessary to increase the intensity of the permitted earthquake hereafter referred to as the "Most Severe Earthquake" (MSE) for the UMAX system. The physical design changes are as follows:

a. Addition of a hold-down system to the Closure Lid to prevent its uplift during MSE.
b. The fill material interstitial space between the CECs, referred to as Space A in Figure 2.4.4, is replaced with plain concrete with a minimum compressive strength of 3000 psi (see Table 2.3.10)
c. The MPC Guides are strengthened to increase their load bearing capacity and their nominal gap with the MPC is engineered to be made vanishingly small.

The design embodiment of HI-STORM UMAX wherein the above changes are incorporated is referred to as "Version MSE."

There are no other upgrades to the design of the "UMAX" ISFSI to meet the loads from the MSE.

1.0.2.1 Change in Design and Applicable Loading

The only significant change contemplated under this CoC Amendment is the introduction of a stronger Design Basis Earthquake (DBE) than that previously approved for the "UMAX" system. The new governing earthquake is quantified in Table 2.3.10 and Table 2.4.5 in Chapter 2.

The seismic analysis to evaluate the admissibility of the more severe DBE is documented in subparagraph 3.4.4.1.2 in Chapter 3.

1.0.2.2 Chapters affected by Amendment 1

The table below lists the locations in this FSAR that contain additions or revisions.

Chapter & Title	Chapter Modified?	Comment/ Justification
1.General Description	Yes	The scope of change applicable to this FSAR revision is summarized in this subsection. The drawing package in Section 1.5 is revised to incorporate the changes mentioned above that together define "Version MSE."
2. Principal Design Criteria	Yes	The intensity of the Design Basis Earthquake is increased as set down in Table 2.3.10. The ISFSI is required to meet the new DBE (termed the Most Severe Earthquake) under five generated earthquakes consisting of statistically independent time histories that all bound the target response spectra per

		SRP3.7.1(see Table 2.4.5).
3. Structural Evaluation	Yes	Additional stress analysis under the so-called Most Severe Earthquake is provided in subparagraph 3.4.4.1.2 in Chapter 3
4. Thermal Evaluation	No	The thermal-hydraulic design of the system is unchanged; hence there is no effect on its thermal characteristics.
5. Shielding Evaluation	No	There is no change in the shielding design features of the system and hence there is no change in its shielding characteristics.
6. Criticality Evaluation	No	There is no change in the MPC or fuel basket design.
7. Confinement Evaluation	No	The design, manufacturing and inspection of the MPC's Enclosure Vessel remains unchanged; therefore, there is no change in the system's confinement capability.
8.Material Evaluation	No	No new material or environmental condition is introduced in this revision.
9.Operating Procedures	No	The loading and unloading procedures remain unchanged.
10. Acceptance Criteria & Maintenance Programs	No	No change
11.Radiation Protection	No	The radiation protection features of the system remain unchanged.
12. Accident Evaluation	No	No new accidents are introduced.
13.Operating Controls and Limits	No	No change in operating controls or limits.
14. Quality Assurance Program	No	The applicable QA program remains unchanged.

As can be seen from the above table, the new text matter and safety evaluation for the proposed change are limited to specific subsections in Chapters 1, 2 and 3. The rest of the FSAR is unaffected by the amendment.

TABLE 1.0.1: APPLICABLE SECTIONS OF HI-STORM FW FSAR*		
Location of UMAX FSAR	**Subject of the Reference**	**Location in HI-STORM FW FSAR, Revision 3**
Section 1.1	Limiting load conditions for MPC-37 and MPC-89	Section 2.2
Sub-Section 1.2.1	Acceptance Criteria for manufacturing of MPCs and HI-TRAC	Section 10.1, Tables 10.1.1, 10.1.3 through 10.1.8
Sub-Section 1.2.1	Description of Alloy X	Appendix 1.A
Sub-Section 1.2.1	Applicable codes for manufacturing of HI-TRAC	Table 1.2.6
Section 1.2.4	1. Overview of loading operations 2. Preparation of HI-TRAC and MPC 3. MPC Fuel Loading 4. MPC Closure 5. Preparation for Storage 6. MPC Inspection Checklist 7. HI-TRAC Inspection Checklist	1. Section 9.2.1 2. Section 9.2.2 3. Section 9.2.3 4. Section 9.2.4 5. Section 9.2.5 6. Table 9.2.4 7. Table 9.2.5

* For convenience of reference, the specific revision of the HI-STORM FW FSAR that is referenced in the safety analysis herein is placed in this docket. Updated versions of the HI-STORM FW FSAR shall be placed in this docket as necessary so as to ensure that the safety analyses on the "UMAX" docket (72-1040) remain aligned with the material referenced in the HI-STORM FW FSAR.

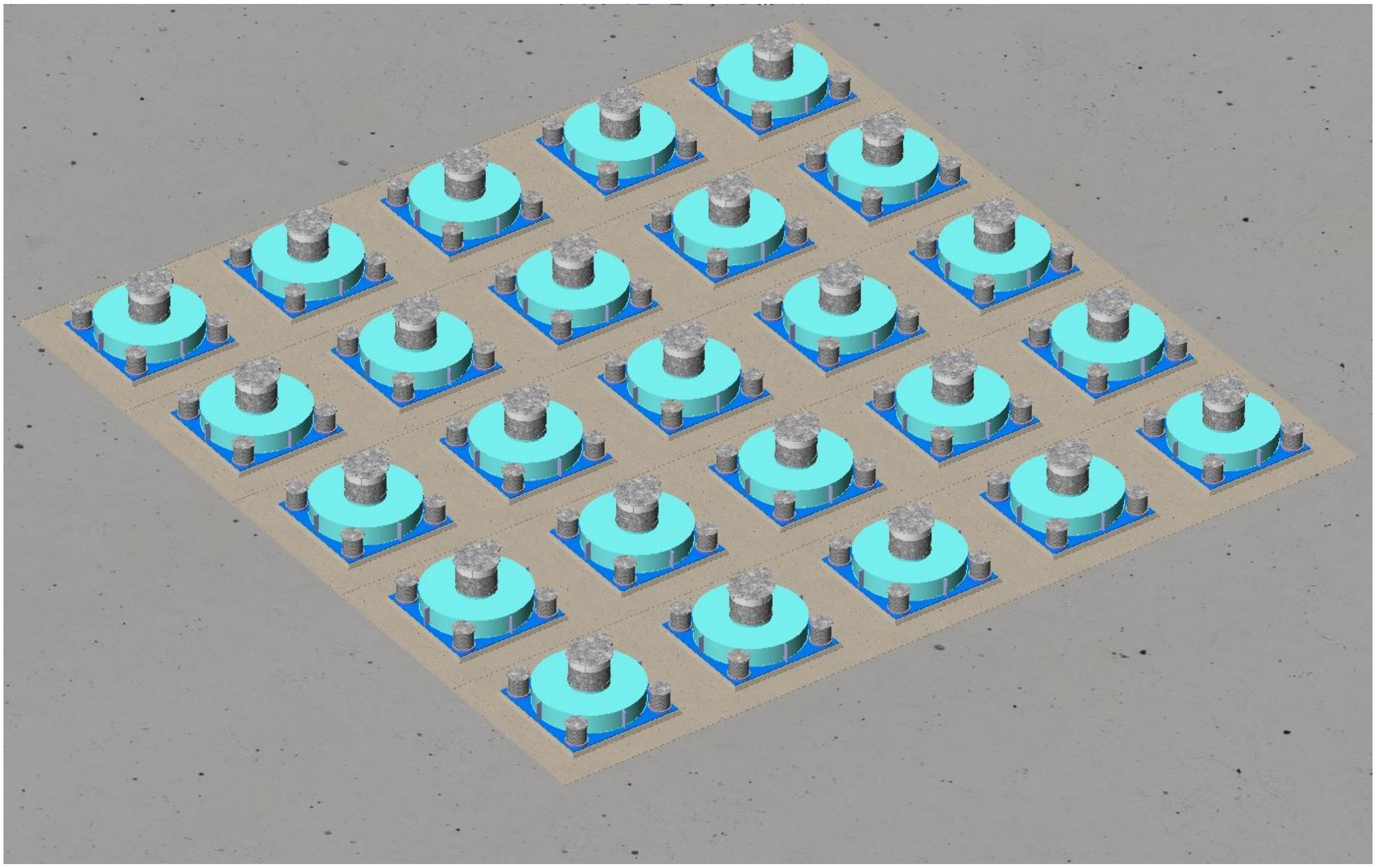


Figure 1.0.1; Pictorial View of a HI-STORM UMAX ISFSI

1.1 INTRODUCTION

HI-STORM UMAX is a dry, in-ground spent fuel storage system consisting of any number of Vertical Ventilated Modules (VVM) each containing one canister. The HI-STORM UMAX is designed to be fully compatible with all HI-TRAC transfer casks and multi-purpose canisters (MPC) presently certified under USNRC Docket No. 72-1014 and 72-1032. Safety analyses documented herein treat all MPCs listed in Table 1.2.1. However, as would be expected, the largest canisters, i.e., those licensed in the HI-STORM FW docket are governing in terms of structural and thermal margins. These largest canisters, namely MPC-37 and MPC-89 are termed "Licensing Basis MPCs" and the certification request for storage in HI-STORM UMAX is limited to these MPCs only. For completeness, the permissible contents from the HI-STORM FW docket are excerpted in Chapter 2 herein and also reproduced in the Technical Specification applicable to the CoC. The safety analyses summarized in this FSAR are intended to demonstrate that the HI-STORM UMAX System can safely store PWR or BWR fuel assemblies, in the MPC-37 or MPC-89, respectively. The MPC is identified by the maximum number of fuel assemblies it can contain in the fuel basket. As presently licensed in the HI-STORM FW docket, the standard MPC external diameters are identical to allow the use of a single overpack design; however the height of the MPC is varied to accord with the SNF to be loaded.

The MPC is an integrally welded pressure vessel designed to meet the stress limits of the ASME Boiler and Pressure Vessel Code, Section III, Subsection NB [1.1.1]. The MPC defines the Confinement Boundary for the stored spent nuclear fuel assemblies. Regardless of the storage cell count, the construction of the MPC is fundamentally the same; the basket is a honeycomb structure comprised of cellular elements. This is positioned within a circumscribing cylindrical canister shell. The egg-crate construction and cell-to-canister shell interface employed in the MPC basket impart the structural stiffness necessary to satisfy the limiting load conditions discussed in Chapter 2 of the HI-STORM FW FSAR. Figures 1.1.1 and 1.1.2 provide cross-sectional views of the PWR and BWR fuel baskets, respectively. Figures 1.1.3 and 1.1.4 provide isometric perspective views of the PWR and BWR fuel baskets, respectively.

The HI-STORM UMAX VVM provides structural protection, cooling, and radiological shielding for the MPC.

The HI-TRAC VW transfer cask (hereafter referred to as HI-TRAC) is required for shielding and protection of the SNF during loading and closure of the MPC and during movement of the loaded MPC from the cask loading area of a nuclear plant spent fuel pool to the storage overpack. Figure 1.1.5 shows a cut away view of the transfer cask. The MPC is placed inside the HI-TRAC transfer cask and moved into the cask loading area of nuclear plant spent fuel pools for fuel loading (or unloading). The HI-TRAC/MPC assembly is designed to prevent (contaminated) pool water from entering the narrow annular space between the HI-TRAC and the MPC while the assembly is submerged. The HI-TRAC transfer cask also allows dry loading (or unloading) of SNF into the MPC in a hot cell.

Design criteria for a forced helium dehydration (FHD) system is provided in Section 2.5.1.

The HI-STORM UMAX is comparable to the HI-STORM 100U ("100U") VVM licensed in NRC Docket No. 72-1014. The major differences between the HI-STORM UMAX and 100U are

that the HI-STORM UMAX VVM cavity is larger in diameter and the HI-STORM UMAX closure lid features a modified outlet ventilation duct system.

The HI-STORM UMAX has all the safety attributes that are attributed to in-ground storage, such as enhanced protection from incident projectiles and threats from extreme environmental phenomena such as hurricanes, tornado borne missiles, earthquakes, tsunamis, fires, and explosions. The HI-STORM UMAX VVM is anatomically similar to the HI-STORM 100U VVM in several respects. In particular, the MPCs are stored in-ground and each storage cavity is isolated from the surrounding environment by a thick cylindrical steel weldment. This steel shell is appropriately coated with surface preservatives or by other means to protect it from corrosion from long-term use.

HI-STORM UMAX differs from HI-STORM 100U in two important respects:

a. The placement of the inlet and outlet ducts in HI-STORM UMAX minimizes the reduction of ventilating air flow rate through HI-STORM VVM under wind conditions which, as is noted in Docket #72-1014, has a small de-rating effect on the thermal performance of the ventilation system in HI-STORM 100U. As shown in Chapter 4, the inlet and outlet locations in HI-STORM UMAX are also found to minimize inter-module flow interactions.

b. The storage cavity in the HI-STORM UMAX VVM is sufficiently large in physical dimensions to accommodate all canisters presently licensed by different designers under different 10CFR72 dockets. Therefore, it is theoretically possible to qualify the entire universe of used fuel canisters presently deployed at the ISFSIs around the country for storage in the HI-STORM UMAX system. This would make it possible to unify the long-term storage of all of the presently deployed nation's dry storage canisters at an Interim Storage site in HI-STORM UMAX VVM assemblies in a monitored and retrievable configuration. The present issue of this SAR, however, is limited to supporting the certification of the HI-STORM FW MPCs listed in Table 1.2.1.

The design and operational attributes of the HI-STORM UMAX, described in the following paragraphs pursuant to the provisions of 10CFR72.24(b), are subject to intellectual property rights in the U.S. and abroad under the patent laws governing the respective jurisdictions.

To summarize, the HI-STORM UMAX System has been engineered to:

- maximize shielding and physical protection for the MPC;
- minimize the extent of handling of the SNF;
- minimize dose to operators during loading and handling;
- require minimal ongoing surveillance and maintenance by plant staff;
- facilitate SNF transfer of the loaded MPC to a compatible transport overpack for transportation;

All HI-STORM UMAX System components (VVM, transfer cask, and MPC) are designated ITS and their sub-components are categorized in accordance with NUREG/CR-6407 [1.1.2].

The principal ancillaries used in the site implementation of the HI-STORM UMAX System are similar to HI-STORM FW System and are summarized in Section 1.2 of the HI-STORM FW FSAR and referenced in Chapter 9 of the HI-STORM FW FSAR in the context of loading operations. A listing of common ancillaries needed by the host site is provided in Table 9.2.1 of the HI-STORM FW FSAR. The detailed design of these ancillaries is not specified in this FSAR. In some cases, there are multiple distinct ancillary designs available for a particular application (such as a forced helium dehydrator or a vacuum drying system for drying the MPC) and as such, not every ancillary will be needed by every site. Ancillary designs are typically specific to a site to meet ALARA and personnel safety objectives.

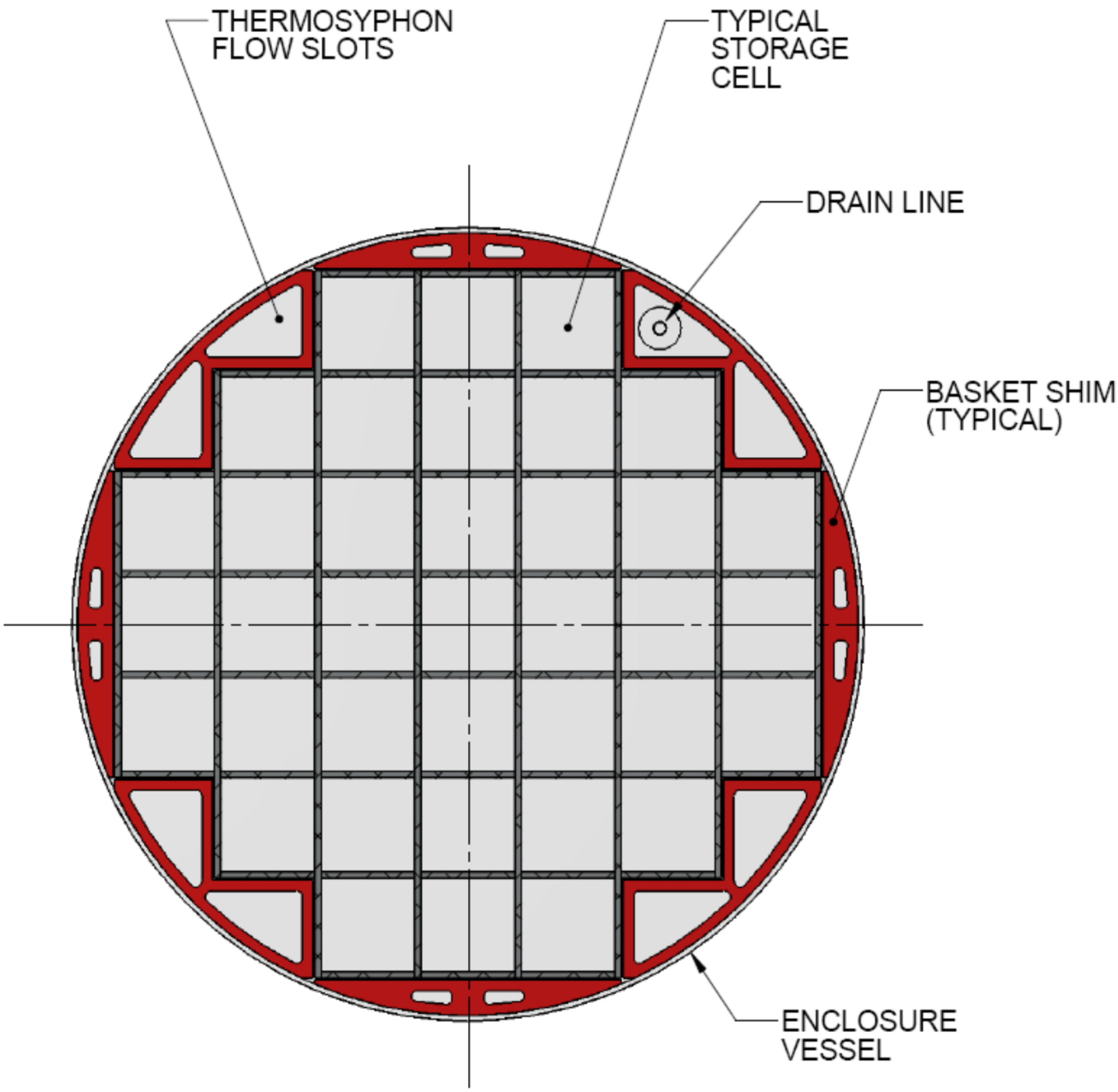


Figure: 1.1.1: MPC-37 in Cross Section

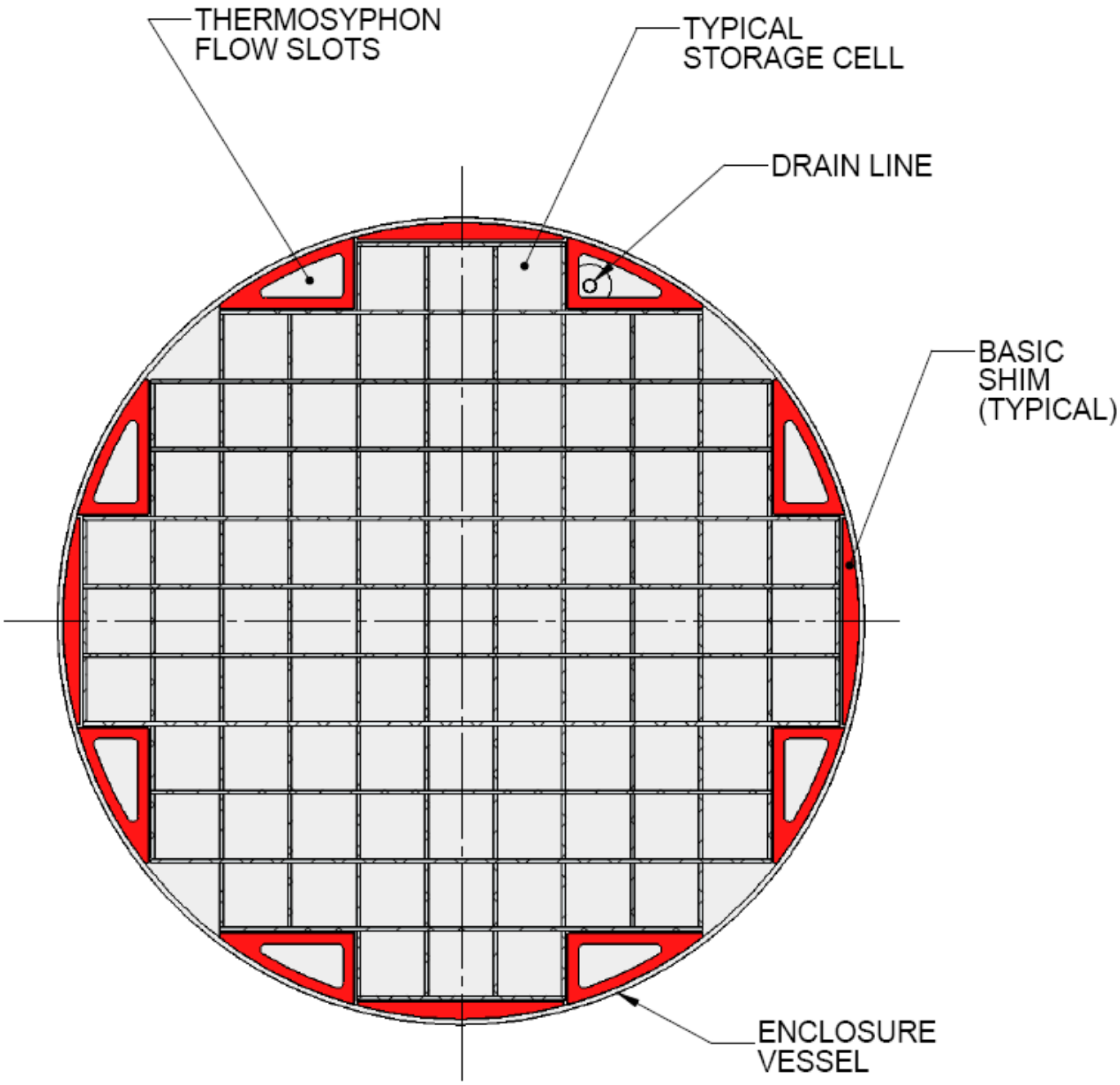


Figure 1.1.2: MPC-89 in Cross Section

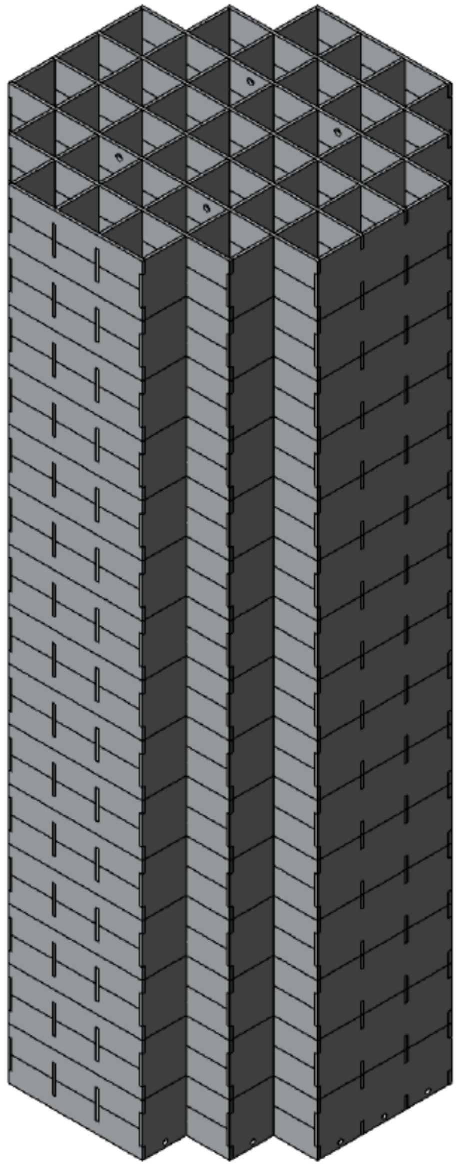


Figure 1.1.3: PWR Fuel Basket (37 Storage Cells) in Perspective View

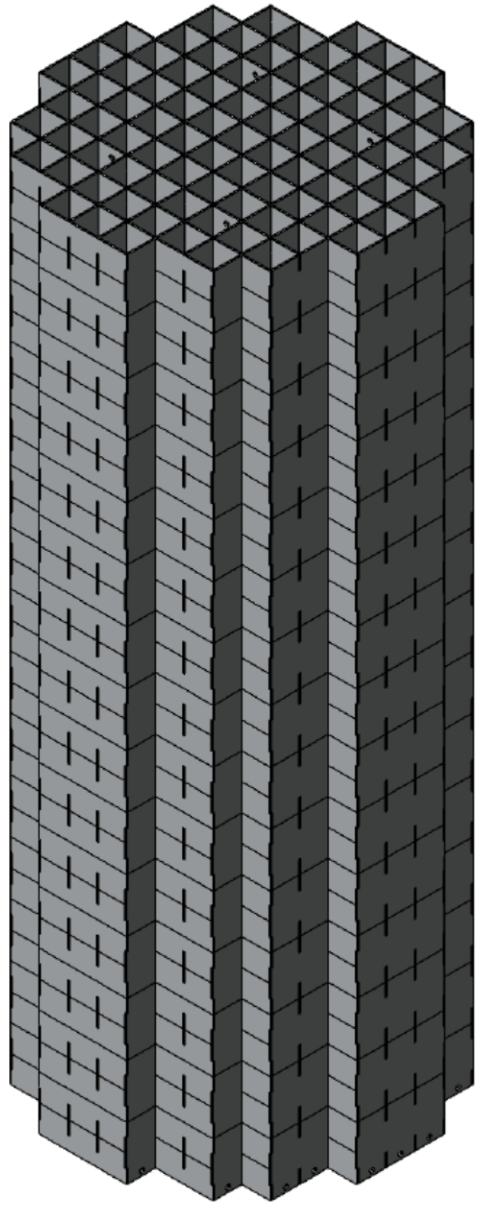


Figure 1.1.4: BWR Fuel Basket (89 Storage Cells) in Perspective View

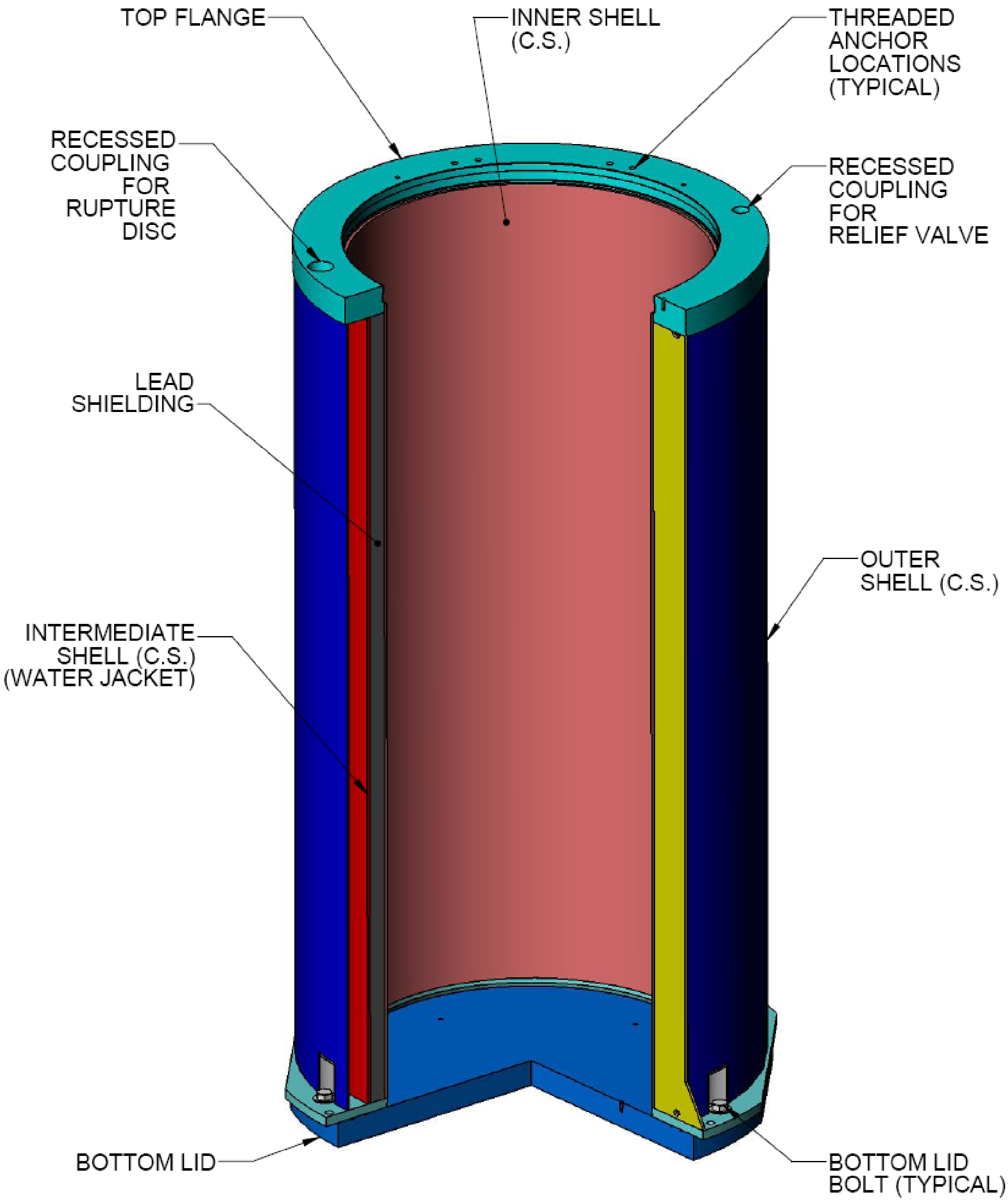


Figure 1.1.5: Cutaway View of HI-TRAC VW

1.2 GENERAL DESCRIPTION OF HI-STORM UMAX SYSTEM

1.2.1 System Characteristics

The HI-STORM UMAX System consists of interchangeable MPCs, which maintain the configuration of the fuel and is the confinement boundary between the stored spent nuclear fuel and the environment; and a storage overpack that provides structural protection and radiation shielding during long-term storage of the MPC. In addition, a transfer cask that provides the structural and radiation protection of an MPC during its loading, unloading, and transfer to the storage overpack is also subject to certification by the USNRC. Description of MPCs and the HI-TRAC transfer cask are provided in Section 1.2 of the HI-STORM FW FSAR. The key parameters for the UMAX MPCs are provided in Table 1.2.2 of the HI-STORM FW FSAR. The principal materials used in the manufacturing of the MPC are listed in the licensing drawings (Section 1.5) and the acceptance criteria are provided in Chapter 10 of HI-STORM FW FSAR. Alloy X description is provided in Appendix 1.A of the HI-STORM FW FSAR. The principal materials used in the manufacturing of the HI-TRAC transfer cask are listed in the licensing drawings in Section 1.5 and the acceptance criteria are provided in Chapter 10 of the HI-STORM FW FSAR. Table 1.2.6 of the HI-STORM FW FSAR provides applicable code paragraphs for manufacturing the HI-TRAC transfer cask.

All structures, systems, and components of the HI-STORM UMAX system, MPCs and HI-TRACs, which are identified as Important-to- Safety (ITS), are specified on the licensing drawings provided in Section 1.5.

1.2.2 Constituents of the HI-STORM UMAX Vertical Ventilated Module and ISFSI Structures

The HI-STORM UMAX VVM, shown in the licensing drawing in Section 1.5, provides for storage of the MPC in a vertical configuration inside a subterranean cylindrical cavity entirely below the top-of-grade (TOG) of the ISFSI. The key constituents of a HI-STORM UMAX VVM and ISFSI structures (see Figure 1.2.1 and Figure 1.2.2) are:

<u>VVM Components</u>

a. The Cavity Enclosure Container (CEC)

b. The Closure Lid

<u>ISFSI Structures</u>

c. The ISFSI Pad

d. The Support Foundation Pad

e. The Subgrade and Under-grade

f. The Enclosure Walls (optional)

A brief description of each constituent part is provided in the following:

a. The Cavity Enclosure Container:

The Cavity Enclosure Container (CEC) consists of a thick walled shell integrally welded to a bottom plate. The top of the container shell is stiffened by a ring shaped flange which is also integrally welded. The constituent parts of the CEC are made of low carbon steel plate, with stainless steel as an optional material for some components, as specified in the licensing drawing provided in Section 1.5. In its installed configuration, the CEC is interfaced with the surrounding subgrade for most of its height except for the top region where it is girdled by the ISFSI pad.

With the Closure Lid removed, the CEC is a closed bottom, open top, thick walled cylindrical vessel that has no penetrations or openings. Thus, groundwater has no path for intrusion into the interior space of the CEC. Likewise, any water that may be introduced into the CEC through the air passages in the top lid will not drain into the groundwater.

The MPC Bearing surfaces and the Divider Shell, two parts internal to the CEC, are important to the thermal performance of the VVM system. The top surfaces of the MPC support system are made of stainless steel so that the MPC is not resting directly on carbon steel components. The Divider Shell, as its name implies, is a vertical cylindrical shell concentrically situated in the CEC that divides the CEC into an inlet flow downcomer and an outlet flow passage. The Divider Shell divides the radial space between the MPC and the CEC cavity into two annuli. The bottom end of the Divider Shell has cutouts to enable movement of air from the downcomer to the up-flow region around the MPC. The cutouts in the Divider Shell are sufficiently tall to ensure that if the cavity were to be filled with water, the bottom region of the MPC would be submerged for several inches. This design feature is important to ensure adequate thermal performance of the system if flood water would stop air flow. The Divider Shell is not attached to the CEC which allows its convenient removal for decommissioning or for any in-service maintenance that may be required.

As the licensing drawing in Section 1.5 shows, the lower end of the MPC is restrained by a set of radial guides at the MPC's Baseplate elevation. The top lid of the MPC is likewise laterally restrained by a set of radial guides attached to the Divider Shell. The radial guides serve as an aid during insertion of the canister into the CEC and also provide the means to limit the lateral movement of the free-standing canisters during an earthquake. By limiting the lateral movement of the MPC, the guides protect against excessive inertia loads during seismic events.

The cylindrical surface of the Divider Shell is equipped with insulation to prevent significant preheating of the inlet air. The insulation material is selected to be water and radiation resistant as well as non-degradable under accidental wetting.

The chief distinguishing features of the VVM are its low profile and in-ground configuration. The MPC is below the ISFSI Pad for its entire height allowing the earth, ISFSI pad and Closure Lid to provide shielding of the stored fuel.

b. The Closure Lid:

The Closure Lid is a steel structure filled with plain concrete that can withstand the impact of the Design Basis Missiles defined in Chapter 2. Both the inlet and outlet vents are located at the grade level. A set of inlet passage located on top of the CEC provide maximum separation from the large outlet passage which is located in the center of the lid. Using a flue extension, the air exhaust from the outlet passage is set to be several feet higher than the inlet and prevents significant preheating of the incoming air. As shown in Chapter 4, the geometry of the inlet and outlet ducts, as depicted in the licensing drawings, make the HI-STORM UMAX VVM essentially insensitive to the speed of the wind.

The Closure Lid fulfills the following principal performance objectives:

1. Because there are no lateral streaming paths in the VVM body and the outlet air passage is located in the Closure Lid, there is no lateral radiation leakage paths during the MPC lowering or raising operation. Thus, the need for shield blocks (necessary to close off vents in certain aboveground HI-STORM 100 models) is eliminated.
2. The air passages in the Closure Lid are configured in such a manner that the aerodynamics in the system is not significantly affected by the change in the horizontal direction of the wind.
3. The Closure Lid is physically restrained against horizontal movement during a Design Basis Earthquake event or a tornado missile strike.
4. To minimize the radiation emitted from the storage cavity, a portion of the Closure Lid extends into the cylindrical space above the MPC. This cylindrical below-surface extension of the Closure Lid is also made of steel filled with shielding concrete to maximize the blockage of skyward radiation issuing from the MPC.
5. As can be seen from the drawings in Section 1.5, the Closure Lid is substantially larger in diameter than the CEC and the MPC is positioned to be at a significant vertical depth below the top of the Container Flange. These geometric provisions ensure that the Closure Lid will not fall into the MPC storage cavity space and strike the MPC if it were accidentally dropped during its handling. Because the Closure Lid is the only removable heavy load, the carefully engineered design features to facilitate recovery from its accidental drop provide added assurance that a handling accident at the ISFSI will not lead to radiological release. This additional measure against accidental Closure Lid drop does not replace the drop prevention features mandated in this FSAR on heavy load lifting devices (such as the cask transporter) that have been a standard and established requirement in the HI-STORM dockets.

c. The ISFSI Pad:

The ISFSI Pad serves to augment shielding, to provide a sufficiently stiff riding surface for the cask transporter, to act as a barrier against gravity-induced seepage of rain or floodwater around the VVM body as well as to shield against a missile. The ISFSI pad is a monolithic reinforced concrete structure that provides the load bearing surface for the cask transporter. The portion of the ISFSI pad adjacent to the VVM is slightly sloped and thicker than the rest of the ISFSI pad to ensure that rain water will be directed away from the VVM. The appropriate requirements on the structural strength of the ISFSI pad and the applicable industry code are specified in Chapter 2.

d. The Support Foundation Pad:

The Support Foundation Pad (SFP) is the underground pad which supports the HI-STORM UMAX ISFSI. The SFP on which the VVM rests must be designed to minimize long-term settlement. The SFP and the under-grade must have sufficient strength to support the weight of all the loaded VVMs during long-term storage and earthquake conditions. As the weight of the loaded VVM is comparable to the weight of the subgrade which it replaces, the additional pressure acting on the SFP is quite small. The appropriate requirements on the structural strength of the SFP and the applicable industry code are specified in Chapter 2.

e. The Subgrade and Under-grade:

The lateral space between each CEC, the SFP and the ISFSI pad is referred to as the subgrade and is filled with a Controlled Low-Strength Material (CLSM). Alternatively, “lean concrete” may also be used.

CLSM is a self-compacted, cementitious material used primarily as a backfill in place of compacted fill. ACI 229R-99 notes several terms, such as flowable fill, unshrinkable fill, controlled density fill, flowable mortar, flowable fly ash, fly ash slurry, plastic soil-cement and soil-cement slurry to describe CLSMs. ACI 116R-00 defines lean concrete as a material with low cementitious content. CLSM and lean concrete are also referred to as “Self-hardening Engineered Subgrade” (SES).

The subgrade material must meet the shear velocity and density requirements in Chapter 2. The space below the SFP is referred to as the under-grade.

As discussed in Chapters 3 and 8 corrosion mitigation measures commensurate with site-specific conditions are implemented on the below grade external surfaces of the CEC.

f. The Enclosure Wall:

The Enclosure Wall is an optional structure which may be utilized to mitigate groundwater intrusion at sites with a high water table.

Analyses in Chapter 3 show that the Self-hardening Engineered Subgrade (SES) provides a stable lateral support system to the ISFSI under the Design Basis Earthquake. In the absence of an Enclosure wall, the interface between the SES and the native subgrade defines the radiation protection boundary of the ISFSI. As stated in the Technical Specification and demonstrated by the seismic analysis in Chapter 3 of this FSAR, excavation of the sub-grade adjacent to the interface of an operating storage system is permitted but only down to the top of the Support Foundation pad elevation.

1.2.3 Design Characteristics of the HI-STORM UMAX VVM

The minimum depth of the storage cavity is governed by the height of the MPC stored. The nominal gap between the top of the MPC and the bottom of the Closure Lid is specified in the

licensing drawing. All Holtec MPC models certified for storage in Docket 72-1032 can be stored in HI-STORM UMAX. The difference in the MPC diameters is accommodated by different sized MPC guides. The location of the MPC Guides on the divider shell is aligned with the bottom of the MPC lid. The pitch between the CEC cavities allows the Cask Transporter to traverse over any storage cavity and independently access any storage location. Thus, any MPC located in any storage cavity can be independently accessed and retrieved using an already certified transfer cask.

As explained in the chapter on operations, the transfer of the MPCs into or out of the storage cavity will occur in an identical manner to HI-STORM 100U using a certified transfer cask. Screens are installed on the air inlet and outlet openings. The flue in the inlet and outlet plenum is equipped with a rain guard. The flue shell is lightweight and fastened to the outlet duct to allow easy installation and removal.

The essential design and operational features of the HI-STORM UMAX System are:

a. Because of its underground staging in HI-STORM UMAX, tip-over of the canister in storage is not possible.

b. To exploit the biological shielding provided by the surrounding soil subgrade, the MPC is entirely situated well below the top-of-grade level. The open plenum above the MPC also acts to boost the ventilation action of the coolant air.

c. Because the VVM is rendered into an integral part of the subgrade, it cannot be translocated to another ISFSI site. It also cannot be lifted and, therefore, is not subject to the potential for a handling accident.

d. Removal of water from the bottom of the storage cavity can be carried out by the simple expedient use of a flexible hose inserted through the air inlet or outlet passageways.

e. As discussed in Section 3.4, all practical efforts are made to coat exposed surfaces of the VVM with proven low VOC and/or ANSI/NSF Standard 61 [1.2.1] compliant surface preservatives to preclude toxicological effects on the environment to the maximum reasonable extent.

1.2.3.1 Shielding Materials

Steel, concrete, and the subgrade are the principal shielding materials in the HI-STORM UMAX. The steel and concrete shielding materials in the Closure lid provide additional gamma and neutron attenuation to reduce dose rates.

Steel, lead, and water are the principal shielding materials in the HI-TRAC transfer cask. The combination of these three shielding materials ensures that the radiation and exposure objectives of 10CFR72.106 and ALARA are met. The extent and location of shielding in the transfer cask plays an important role in minimizing the personnel doses during loading, handling, and transfer.

The MPC fuel basket structure provides the initial attenuation of gamma and neutron radiation emitted by the radioactive contents. The MPC shell, baseplate, and thick lid provide additional gamma attenuation to reduce direct radiation.

1.2.3.1.1 Neutron Absorber – Metamic HT

Metamic-HT is the designated neutron absorber in the HI-STORM FW MPC baskets. It is also the structural material of the basket. The properties of Metamic-HT and key characteristics, necessary for ensuring nuclear reactivity control, thermal, and structural performance of the basket, are presented in Metamic-HT Sourcebook [1.2.4].

1.2.3.1.2 Neutron Shielding

The HI-TRAC transfer cask is equipped with a water jacket providing radial neutron shielding. The water in the water jacket may be fortified with ethylene glycol to prevent freezing under low temperature operations [1.2.3].

During certain evolutions in the short term handling operations, the MPC may contain water which will supplement neutron shielding.

1.2.3.1.3 Gamma Shielding

In the HI-TRAC transfer cask, the primary gamma shielding is provided by lead. As in the storage overpack, carbon steel supplements the lead gamma shielding of the HI-TRAC transfer cask.

In the MPC, the gamma shielding is provided by its stainless steel enclosure vessel (including a thick lid); and its aluminum based fuel basket and aluminum alloy basket shims.

1.2.3.2 Lifting Devices

Lifting and handling devices used to load or unload an MPC into the HI-STORM UMAX VVM shall be designed per Section 1.2.1 of the HI-STORM FW FSAR (placed in this docket).

The lifting and handling of all heavy loads that are within Part 72 jurisdiction, such as the Closure Lid, to be carried out using single failure proof (see definition in the Glossary) equipment with below-the-hook lifting devices that comply with the stress limits of ANSI N14.6 [1.2.2] and/or slings designed as single failure proof to render an uncontrolled lowering of the Lid a non-credible event.

1.2.3.3 Threaded Anchor Locations

Threaded anchor locations (TALs) are provided in the CEC Flange region of each CEC. These will serve as the anchoring location for the device used for MPC transfer (see Chapter 9). The TALs serve no function during long term storage.

1.2.3.4 Design Life

The design life of the HI-STORM UMAX System is 60 years. This is accomplished by using materials of construction with a long proven history in the nuclear industry, specifying materials known to withstand their operating environments with little to no degradation (see Chapter 8), and protecting material from corrosion by using appropriate mitigation measures.

A maintenance program, as specified in Chapter 10, is also implemented to ensure that the service life will exceed the design life. The design considerations that assure the HI-STORM UMAX System performs as designed include the following:

<u>HI-STORM UMAX VVM and HI-TRAC Transfer Cask</u>

a. Exposure to Environmental Effects
b. Material Degradation
c. Maintenance and Inspection Provisions

<u>MPCs</u>

a. Corrosion
b. Structural Fatigue Effects
c. Maintenance of Helium Atmosphere
d. Allowable Fuel Cladding Temperatures
e. Neutron Absorber Boron Depletion

The adequacy of the materials for the designated design life is discussed in Chapter 8.

1.2.4 Operational Characteristics of the HI-STORM UMAX

Fuel loading operations, MPC preparation, and requirements during the use of the transfer cask are described in the HI-STORM FW Final Safety Analysis Report. The HI-TRAC transfer cask is used for on-site transport of the loaded MPC from the Fuel Building to the ISFSI. Prior to loading the VVM, the Closure Lid or other temporary lid is removed and a suitable device which will connect the transfer cask to the CEC is installed. The cask transporter carrying the transfer cask with the loaded MPC aligns over the top of the CEC and the transfer cask is connected. The MPC inside the transfer cask is lifted slightly by the cask transporter to allow the transfer cask pool lid to be removed. The MPC is slowly lowered into the VVM cavity below. The transfer equipment is removed and the Closure Lid is installed. The principal operational characteristics of short term operations at an ISFSI are:

a. The MPC is kept in the vertical configuration at all times during handling operations. This eliminates the handling risk of down-ending or upending and maintains the thermal performance of the MPC (which is somewhat dependent on its orientation) undisturbed.

b. The vertical insertion (or withdrawal) of the MPC eliminates the risk of gouging or binding of the MPC with the CEC parts.

c. The HI-TRAC transfer cask is mounted on the VVM cavity (with an interposed Mating Device) with large fasteners that are sized to protect the transfer cask from tip over under the site's DBE.

d. All load handling operations are carried out using the Vertical Cask Transporter (VCT) or an equivalent crane that is *single failure proof.*

Details of the generic operational steps involving either installation or removal of the loaded MPC at a HI-STORM UMAX ISFSI are provided in Chapter 9 along with reference to certain recommended safety measures that are known from experience to avert human performance errors. A visual depiction of the required operational steps is also provided in Chapter 9.

1.2.4.1 Design Features

The design features of the HI-STORM UMAX System are intended to meet the following principal performance characteristics under all credible modes of operation:

a. Prevent unacceptable release of contained radioactive material at all times.

b. Minimize occupational and site boundary dose.

c. Permit retrievability of contents (fuel must be retrievable from the MPC under normal and off-normal conditions in accordance with ISG-2 and the MPC only must be recoverable after accident conditions in accordance with ISG-3).

Chapter 11 identifies the many design features built into the HI-STORM UMAX System to minimize dose and maximize personnel safety. Among the design features intrinsic to the system that facilitate meeting the above objectives are:

a. The loaded MPC is always maintained in a vertical orientation during its handling at the ISFSI and is handled using an ANSI N14.6 compliant lift cleats.

b. The height of the HI-STORM UMAX cavity is minimized consistent with the length of the MPCs to optimize the depth of excavation needed to establish the ISFSI.

c. Almost all personnel activities during MPC transfer occur at ground level which helps promote safety and ALARA.

1.2.4.2 Identification of Subjects for Safety and Reliability Analysis

1.2.4.2.1 Criticality Prevention

Criticality is controlled by geometry and neutron absorbing materials in the fuel basket. The entire basket is made of Metamic-HT, a uniform dispersoid of boron carbide and nano-particles of alumina in an aluminum matrix, serves as the neutron absorber. This accrues four major safety and reliability advantages:

(i) The larger B-10 areal density in the Metamic-HT allows higher enriched fuel (i.e., BWR fuel with planar average initial enrichments greater than 4.5 wt% U-235) without relying on gadolinium or burn-up credit.

(ii) The neutron absorber cannot be removed from the basket or displaced within it.

(iii) Axial movement of the fuel with respect to the basket has no reactivity consequence because the entire length of the basket contains the B-10 isotope.

(iv) The larger B-10 areal density in the Metamic-HT reduces the reliance on soluble boron credit during loading/unloading of PWR fuel.

1.2.4.1.2 Chemical Safety

There are no chemical safety hazards associated with operations of the HI-STORM UMAX System. A detailed evaluation is provided in Section 3.4.

1.2.4.2.3 Operation Shutdown Modes

The HI-STORM UMAX System is totally passive and consequently, operation shutdown modes are unnecessary.

1.2.4.2.4 Instrumentation

As stated earlier, the HI-STORM UMAX MPC, which is seal welded, non-destructively examined, and pressure tested, confines the radioactive contents. The HI-STORM UMAX is a completely passive system with appropriate margins of safety; therefore, it is not necessary to deploy any instrumentation to monitor the cask in the storage mode.

1.2.4.2.5 Maintenance Technique

Because of its passive nature, the HI-STORM UMAX System requires minimal maintenance over its lifetime. No special maintenance program is required. Chapter 10 describes the maintenance program set forth for the HI-STORM UMAX System.

1.2.5 Cask Contents

This sub-section contains information on the cask contents pursuant to 10 CFR72, paragraphs 72.2(a)(1),(b) and 72.236(a),(c),(h),(m).

The HI-STORM UMAX System is designed to house both BWR and PWR spent nuclear fuel assemblies. Table 1.2.3 provides key system data and parameters for the MPCs. A description of acceptable fuel assemblies for storage in the MPCs is provided in Section 2.1. This includes fuel assemblies classified as damaged fuel assemblies and fuel debris in accordance with the definitions of these terms in the Glossary. All fuel assemblies, non-fuel hardware, and neutron sources authorized for packaging in the MPCs must meet the fuel specifications provided in Section 2.1. All fuel assemblies classified as damaged fuel or fuel debris must be stored in damaged fuel containers (DFC).

As shown in Figure 2.1.7 (MPC-37) and Figure 2.1.8 (MPC-89), each storage location is assigned an associated cell identification number. A DFC can be stored in the outer peripheral locations of both MPC-37 and MPC-89 as shown in Figures 2.1.1 and 2.1.2, respectively. The permissible heat loads for each cell, and the total canister are given in Tables 2.1.8 and 2.1.9 for MPC-37 and MPC-89, respectively.

Table 1.2.1

MULTI-PURPOSE CANISTER MODELS GEOMETRICALLY COMPATIBLE FOR STORAGE IN HI-STORM UMAX

USNRC Docket Number	**MPC Model I.D.(Note 1)**
72-1014	MPC-32/32F
	MPC-24
	MPC-24E/24EF
	MPC-68/68FF
	MPC-68F
	MPC-68M
72-1032	MPC-37
	MPC-89

Note 1: This issue of the FSAR seeks certification of only MPC-89 and MPC-37, termed as “licensing basis canisters” for HI-STORM UMAX. The analyses performed on other MPCs are for reference purposes only.

Table 1.2.2

HI-TRAC MODELS IN HI-STORM DOCKETS CORRESPONDING TO THE MPCs IN TABLE 1.2.1

USNRC Docket Number	**Model ID (Note 1)**
72-1014	HI-TRAC 100
	HI-TRAC 100D
	HI-TRAC 125
	HI-TRAC 125D
72-1032	HI-TRAC VW

Note 1: This issue of the FSAR seeks certification of only MPC-89 and MPC-37, termed as "licensing basis canisters" for HI-STORM UMAX. The analyses performed on other MPCs are for reference purposes only.

Table 1.2.3

KEY SYSTEM DATA FOR HI-STORM UMAX SYSTEM

ITEM	QUANTITY	NOTES
Types of MPCs	2	1 for PWR 1 for BWR
MPC storage capacity†:	MPC-37	Up to 37 undamaged ZR clad PWR fuel assemblies with or without non-fuel hardware. Up to 12 damaged fuel containers containing PWR damaged fuel and/or fuel debris may be stored in the locations denoted in Figure 2.1.1 with the remaining basket cells containing undamaged fuel assemblies, up to a total of 37.
MPC storage capacity†:	MPC-89	Up to 89 undamaged ZR clad BWR fuel assemblies. Up to 16 damaged fuel containers containing BWR damaged fuel and/or fuel debris may be stored in locations denoted in Figure 2.1.2 with the remaining basket cells containing undamaged fuel assemblies, up to a total of 89.

† See Chapter 2 for a complete description of authorized cask contents and fuel specifications.

[PROPRIETARY INFORMATION WITHHELD IN ACCORDANCE WITH 10 CFR 2.390]

Figure 1.2.1 – Key Constituent Parts of a HI-STORM UMAX VVM

Note: The design features of the HI-STORM UMAX System are the exclusive intellectual property of Holtec International under U.S. and international patent right laws. Minor details of the HI-STORM UMAX depicted here may vary slightly from the licensing drawings in Section 1.5. This figure is for illustrative purposes only.

[PROPRIETARY INFORMATION WITHHELD IN ACCORDANCE WITH 10 CFR 2.390]

Figure 1.2.2 – HI-STORM UMAX VVM Shown within the ISFSI Structure

Note: The design features of the HI-STORM UMAX System are the exclusive intellectual property of Holtec International under U.S. and international patent right laws. Minor details of the HI-STORM UMAX depicted here may vary slightly from the licensing drawings in Section 1.5. This figure is for illustrative purposes only.

1.3 IDENTIFICATION OF AGENTS AND CONTRACTORS

This section contains the necessary information to fulfill the requirements pertaining to the qualifications of the applicant pursuant to 10 CFR72.2(a)(1),(b) and 72.230(a). Holtec International, based in Marlton, NJ, is the system designer and applicant for certification of the HI-STORM UMAX system.

Holtec International is an engineering technology company with a principal focus on the power industry. Holtec International Nuclear Power Division (NPD) specializes in spent fuel storage technologies. NPD has carried out turnkey wet storage capacity expansions (engineering, licensing, fabrication, removal of existing racks, performance of underwater modifications, volume reduction of the old racks and hardware, installation of new racks, and commissioning of the fuel pool for increased storage capacity) in numerous nuclear plants around the world. Over 90 plants in the U.S., Britain, Brazil, Korea, Mexico, China and Taiwan have utilized the Company's wet storage technology to establish their state-of-the-art in-pool storage capacities.

Holtec's NPD is also a turnkey provider of dry storage and transportation technologies to nuclear plants around the globe. The company is contracted by over 45 nuclear units in the U.S. to provide the company's vertical ventilated dry storage technology. Utilities in China, Korea, Spain, Ukraine, the United Kingdom and Switzerland are also active users of Holtec International's dry storage and transport systems.

Four U.S. commercial plants, namely, Dresden Unit 1, Trojan, Indian Point Unit 1, and Humboldt Bay have thus far been completely defueled using Holtec International's technology. For many of its dry storage clients, Holtec International provides all phases of dry storage including: the required site-specific safety evaluations; ancillary designs; manufacturing of all capital equipment; preparation of site construction procedures; personnel training; dry runs; and fuel loading. The USNRC dockets in parts 71 and 72 currently maintained by the Company are listed in Table 1.3.1.

Holtec International's corporate engineering consists of professional engineers and experts with extensive experience in every discipline germane to the fuel storage technologies, namely structural mechanics, heat transfer, computational fluid dynamics, and nuclear physics. Virtually all engineering analyses for Holtec's fuel storage projects (including HI-STORM UMAX) are carried out by the company's full-time staff. The Company is actively engaged in a continuous improvement program of the state-of-the-art in dry storage and transport of spent nuclear fuel. The active patents and patent applications in the areas of dry storage and transport of SNF held by the Company (ca. January 2012) are listed in Table 1.3.2. Table 1.3.3 lists Holtec patents on dry storage technologies that have been published by the US patent office (as of Jan .2012) but not yet granted. Many of these listed patents have been utilized in the design of the HI-STORM UMAX System.

Holtec International's quality assurance (QA) program was originally developed to meet NRC requirements delineated in 10CFR50 [1.3.1], Appendix B, and was expanded to include provisions of 10CFR71 [1.3.2], Subpart H, and 10CFR72, Subpart G, for structures, systems, and components designated as important to safety. The Holtec quality assurance program, which satisfies all 18 criteria in 10CFR72, Subpart G, that apply to the design, fabrication, construction, testing, operation, modification, and decommissioning of structures, systems, and components

important to safety is incorporated by reference into this FSAR. Holtec International's QA program has been certified by the USNRC (Certificate No. 71-0784).

The HI-STORM UMAX System will be fabricated by the Holtec International Manufacturing Division (HMD) located in Pittsburgh, Pennsylvania with subcontract services from the Company's smaller plants in other parts of the country, namely Nanotec Metals Division (NMD) in Lakeland, FL and Orrvilon in Orrville, Ohio. HMD is a long-term ASME N-Stamp holder and fabricator of nuclear components. In particular, HMD has been manufacturing HI-STORM and HI-STAR system components since the inception of Holtec International's dry storage and transportation program in the 1990s. HMD routinely manufactures ASME code components for use in the U.S. and overseas nuclear plants. Holtec International's engineering organization, based in Marlton, NJ and the HMD subsidiary in Pittsburgh, PA have been subject to triennial inspections by the USNRC. If another fabricator is to be used for the fabrication of any part of the HI-STORM UMAX System, the proposed fabricator will be evaluated and audited in accordance with Holtec International's QA program approved by the USNRC.

Holtec International's Nuclear Power Division (NPD) also carries out site services for dry storage deployments at nuclear power plants. Several nuclear plants, such as Trojan and Waterford 3 (both completed) and Comanche Peak (ongoing, ca. January 2012) have deployed dry storage at their sites using a turnkey contract with Holtec International.

Table 1.3.1

USNRC DOCKETS ASSIGNED TO HOLTEC INTERNATIONAL

System Name	Docket Number
HI-STORM 100 (Storage)	72-1014
HI-STAR 100 (Storage)	72-1008
HI-STAR 100 (Transportation)	71-9261
HI-STAR 180 (Transportation)	71-9325
HI-STAR 60 (Transportation)	71-9336
Holtec Quality Assurance Program	71-0784
HI-STORM FW (Storage)	72-1032
HI-STORM UMAX	72-1040

Table 1.3.2
DRY STORAGE AND TRANSPORT PATENTS HELD BY HOLTEC INTERNATIONAL

Item No.	Colloquial Name of the Patent	USPTO Patent Number
1.	Honeycomb Fuel Basket	5,898,747
2.	Radiation Absorbing Refractory Composition (METAMIC)	5,965,829
3.	HI-STORM 100S Overpack	6,064,710
4.	Extrusion Fabrication Process for Discontinuous Carbide Particulate Metal Matrix Composites and Super Hypereutectic A1/S1(METAMIC-CLASSIC)	6,042,779
5.	Duct Photon Attenuator	6,519,307B1
6.	HI-TRAC Operation	6,587,536B1
7.	Cask Mating Device (Hermetically Sealable Transfer Cask)	6,625,246B1
8.	Improved Ventilator Overpack	6,718,000B2
9.	Below Grade Transfer Facility	6,793,450B2
10.	HERMIT (Seismic Cask Stabilization Device)	6,848,223B2
11.	Cask Mating Device (operation)	6,853,697
12.	Davit Crane	6,957,942B2
13.	Duct-Fed Underground HI-STORM	7,068,748B2
14.	Forced Helium Dehydrator (design)	7,096,600B2
15.	Below Grade Cask Transfer Facility	7,139,358B2
16.	Forced Gas Flow Canister Dehydration (alternate embodiment)	7,210,247B2
17.	HI-TRAC Operation (Maximizing Radiation Shielding During Cask Transfer Procedures)	7,330,525
18.	HI-STORM 100U	7,330,526B2
19.	Flood Resistant HI-STORM	7,590,213B1
20.	HI-STORM 100M (Underground Manifolded module assembly)	7,676,016B2
21.	Dew Point Temperature Based Canister Dehydration	7,707,741B2
22.	Optimized Weight Transfer Cask with Detachable Shielding	7,786,456B2
23.	VESCAP (Apparatus, System, and Method for Facilitating Transfer of High Level Radioactive Waste to and/or From a Pool	7,820,870B2
24.	HI-STORM 100F (Counter-flow Underground Vertical Ventilated Module)	7,933,374B2
25.	Apparatus for Transporting and/or Storing Radioactive Materials Having Jacket Adapted to Facilitate Thermo-siphon Fluid Flow	7,994,380B2
26.	Method of Removing Radioactive Materials from Submerged State and/or Preparing Spent Nuclear Fuel for Dry Storage	8,067,659B2
27.	HI-STORM 100US	8,098,790B1
28.	Canister Apparatus and Basket for Transporting, Storing and/or Supporting Spent Nuclear Fuel(Double Wall Canister)	8,135,107B2

Table 1.3.3

HOLTEC INTERNATIONAL PENDING PATENTS ON FUEL STORAGE

	Title	Submittal Date	*USPTO FILE NUMBER*	
1.	System And Method For The Ventilated Storage Of High Level Radioactive Waste In A Clustered Arrangement(HIC-Storm)	22-Dec-08	*12340948*	US20090159550
2.	System And Method For Preparing A Container Loaded With Wet Radioactive Elements For Dry Storage(Inter-Unit Transfer)	22-Dec-08	*12342022*	US20090158614
3.	Apparatus And Method For Supporting Fuel Assemblies In An Underwater Environment Having Lateral Access Loading	31-Mar-10	*12751717*	US20100232563
4.	Neutron Shielding Ring, Apparatus And Method Using The Same For Storing High Level Radioactive Waste (HI-STAR 180)	02-Jul-07	*11772581*	US20080084958
5.	Fuel Basket Spacer, Apparatus And Method Using The Same For Storing High Level Radioactive Waste (HI-STAR 180)	02-Jul-07	*11772620*	US20080031397
6.	Spent Fuel Basket, Apparatus And Method Using The Same For Storing High Level Radioactive Waste (HI-STAR 180)	02-Jul-07	*11772610*	US20080031396
7.	Method And Apparatus For Dehydrating High Level Waste Based On Dew Point Temperature Measurements (FGD)	18-Dec-09	*12641378*	US20100212182
8.	System And Method For Storing Spent Nuclear Fuel Having Manifolded Underground Vertical Ventilated Module (100M)	19-Feb-10	*12709094*	US20100150297

Table 1.3.3

HOLTEC INTERNATIONAL PENDING PATENTS ON FUEL STORAGE

	Title	**Submittal Date**	***USPTO FILE NUMBER***	
9.	Cask Apparatus, System And Method For Transporting And/Or Storing High Level Waste (HI-SAFE)	28-Apr-10	*12769622*	US20100272225
10.	Spent Fuel Basket For Storing High Level Radioactive Waste (HEXCOMB Racks)	29-Oct-08	*12260914*	US20090175404
11.	System and Method for Reclaiming Energy from Heat Emanating from Spent Nuclear Fuel	21-Apr-11	*13092143*	US20110286567
12.	Method Of Storing High Level Waste (100F)	26-Apr-11	*13094498*	US20110255647
13.	Single-Plate Neutron Absorbing Apparatus And Method Of Manufacturing The Same	23-Dec-09	*12645846*	US20110033019
14.	Apparatus For Storing And/Or Transporting High Level Radioactive Waste, And Method For Manufacturing The Same	06-May-10	*12774944*	US20100284506
15.	System, Method And Apparatus For Providing Additional Radiation Shielding To High Level Radioactive Materials	05-Nov-10	*12940804*	US20110172484
16.	Atomized Pico-scale Composite Aluminum Alloy And Method Thereof	24-Apr-09	*12312089*	US20100028193

1.4 GENERIC CASK ARRAYS

An ISFSI deploying the HI-STORM UMAX System may use an unlimited number of VVM. The preferred embodiment of the VVM array is a rectangular configuration as illustrated in the licensing drawing in Section 1.5. The reference pitch (center-to-center distance between centerlines of adjacent CECs in the two orthogonal directions) between the VVMs is shown on the licensing drawing. In either or both directions, the reference pitch spacing can be adjusted by the site to ensure that any commercially available cask transporters can traverse the VVM arrays to provide autonomous access to each stored MPC. This reference pitch spacing also serves to provide adequate shielding around each storage cavity.

No limit is placed on the maximum spacing between VVMs. If there are multiple VVMs in an ISFSI they should be founded on a continuous SFP, to the extent practicable, to enhance the seismic response characteristics of the ISFSI.

1.5 FIGURES AND DRAWINGS

The licensing drawing for the HI-STORM UMAX System, pursuant to the requirements of 10CFR72.24(c)(3), is provided in this section. The material list on the licensing drawing contains sufficient information to articulate major design features and general operational characteristics of "UMAX". Further, it is intended to serve as the control information to guide the preparation of the documents required to manufacture the components under Holtec's Quality Assurance Program. Holtec's Quality Assurance Program requires that the entire array of manufacturing documents must remain in complete conformance with the Licensing Drawing Package at all times.

The MPC and HI-TRAC drawings listed below are excerpted from the HI-STORM FW docket.

Drawing Package Number	Description	Revision
8446	HI-STORM UMAX Canister Storage System	15
6514	HI-TRAC VW – MPC-37	9
6799	HI-TRAC VW – MPC-89	9
6505	MPC-37 ENCLOSURE VESSEL	15
6506	MPC-37 FUEL BASKET	12
6512	MPC-89 ENCLOSURE VESSEL	17
6507	MPC-89 FUEL BASKET	11

[PROPRIETARY DRAWINGS WITHHELD IN ACCORDANCE WITH 10 CFR 2.390]

1.6 REFERENCES

[1.0.1] USNRC Docket 72-1014, "Final Safety Analysis Report for the HI-STORM 100 Cask System", Holtec Report No. HI-2002444, latest revision.

[1.0.2] "Final Safety Analysis Report on the HI-STORM FW System", Holtec Report No. HI-2114830, latest revision.

[1.0.3] 10 CFR Part 72, "Licensing Requirements for the Independent Storage of Spent Nuclear Fuel, High-level Radioactive Waste, and Reactor-Related Greater than Class C Waste", Title 10 of the Code of Federal Regulations- Energy, Office of the Federal Register, Washington, D.C.

[1.0.4] USNRC Regulatory Guide 3.61 (Task CE.306-4) "Standard Format for a Topical Safety Analysis Report for a Spent Fuel Dry Storage Cask", February 1989.

[1.0.5] USNRC NUREG-1536, "Standard Review Plan for Spent Fuel Dry Storage Systems at a General License Facility", Revision 1, July 2010.

[1.1.1] ASME Boiler & Pressure Vessel Code, Section III, Subsection NB, American Society of Mechanical Engineers, New York, 2010.

[1.1.2] NUREG/CR-6407, "Classification of Transportation Packaging and Dry Spent Fuel Storage System Components According to Importance to Safety", U.S. Nuclear Regulatory Commission, February 1996.

[1.2.1] ANSI/NSF Standard 61, "Drinking Water System Components – Health Effects".

[1.2.2] ANSI N14.6-1993, "American National Standard for Radioactive Materials - Special Lifting Devices for Shipping Containers Weighing 10,000 Pounds (4,500 Kg) or More", American National Standards Institute, Inc., Washington D.C., June 1993.

[1.2.3] Companion Guide to the ASME Boiler & Pressure Vessel Code, K.R. Rao (editor), Chapter 56, " Management of Spent Nuclear Fuel", Third Edition, ASME (2009).

[1.2.4] Metamic-HT Qualification Sourcebook, Holtec Report HI-2084122, Latest Revision.

[1.3.1] 10CFR Part 50, "Domestic Licensing of Production and Utilization Facilities", Title 10 of the Code of Federal Regulations, Office of the Federal Register, Washington, D.C.

[1.3.2] 10CFR Part 71, "Packaging and Transportation of Radioactive Material", Title 10 of the Code of Federal Regulations, Office of the Federal Register, Washington, D.C.

CHAPTER 2: PRINCIPAL DESIGN CRITERIA

2.0 OVERVIEW OF THE PRINCIPAL DESIGN CRITERIA

2.0.1 General

This chapter provides a systematic presentation of the loadings that must be considered for a complete safety evaluation of the HI-STORM UMAX System. As discussed in Chapter 1, this FSAR does not introduce any new MPC or transfer cask model. Rather, it envisages previously approved MPCs under docket 72-1032 [2.0.1] to be stored in a HI-STORM UMAX VVM. The drawings of the previously certified MPC-37 and MPC-89 are reproduced from the HI-STORM FW docket in Section 1.5 herein. Although the safety analyses of these previously certified MPCs are adopted from the HI-STORM FW FSAR by reference, additional safety evaluations are necessary to ensure that their licensed limits set forth in this CoC are not exceeded when they are stored in the HI-STORM UMAX VVM. Likewise, it is necessary to ensure that the licensed limits of the HI-TRAC VW transfer cask, certified along with the MPC-37 and MPC-89 in docket number 72-1032, are not exceeded when it is used to transfer the MPC at the HI-STORM UMAX ISFSI.

Section 1.5 contains the Licensing drawings for the MPC-37, MPC-89 and HI-TRAC VW cask excerpted from the HI-STORM FW docket where it was originally certified.

Design Criteria pertaining to the loadings and components common to the HI-STORM FW and the HI-STORM UMAX systems, such as the MPC and the HI-TRAC, are referenced in this FSAR, as appropriate, to the HI-STORM FW FSAR. To facilitate convenient access to the referenced material, the latest edition of the HI-STORM FW FSAR has been placed in this docket and a list of HI-STORM FW FSAR sections germane to this chapter is provided in a tabular form below.

HI-STORM FW FSAR material referenced in this FSAR		
Location of UMAX FSAR	**Subject of the Reference**	**Location in HI-STORM FW FSAR, Revision 3**
Sub-Section 2.0.1	Design Characteristics of MPC	Sub-Section 1.2.1
Sub-Section 2.0.1	Design Characteristics of HI-TRAC	Paragraph 1.2.1.3
Sub-Section 2.0.2	Structural analysis for qualifying the MPC closure welds	Sub-Section 3.4.3 and sub-Section 3.4.4
Sub-Section 2.0.2	MPC design analyzed for all design basis normal, off-normal and postulated accident conditions	Sub-Sections 2.2.1, 2.2.2, and 2.2.3
Sub-Section 2.0.2	HI-TRAC design analyzed for all design basis normal, off-normal and postulated accident conditions	Sub-Sections 2.2.1, 2.2.2, and 2.2.3
Sub-Section 2.0.3	HI-TRAC designed for off-normal environmental cold conditions	Section 2.2.2
Sub-Section 2.0.3	Evaluation of potential for brittle fracture in structural steel materials	Sub-Section 3.1.2.4 and Table 3.1.9
Sub-Section 2.0.4	HI-TRAC accident condition- loss of water in the water jacket	Sub-Section 5.1.2
Sub-Section 2.0.5	MPC providing criticality control for all design basis normal, off-normal and postulated accident conditions	Section 6.1
Sub-Section 2.0.6	MPC meeting the guidance of ISG-18	Section 7.1
Sub-Section 2.0.8	Acceptance Criteria for manufacturing of MPCs and HI-TRAC	Section 10.1, Tables 10.1.1, 10.1.3 through 10.1.8
Sub-Section 2.0.8	Maintenance Program for MPCs and HI-TRAC	Section 10.2, Table 10.2.1
Sub-Section 2.3.1	Applicable loads to MPC and HI-TRAC	Tables 2.2.6, 2.2.7, 2.2.13 and 3.1.1

HI-STORM FW FSAR material referenced in this FSAR*		
Location of UMAX FSAR	**Subject of the Reference**	**Location in HI-STORM FW FSAR, Revision 3**
Subsection 2.3.4	Confinement Boundary Leakage	Section 7.1
Sub-Section 2.7.2	Inspections and tests to verify integrity of the confinement boundary	Sub-Section 10.1.4

* For convenience of reference, the specific revision of the HI-STORM FW FSAR that is referenced in the safety analysis herein is placed in this docket. Updated versions of the HI-STORM FW FSAR shall be placed in this docket as necessary so as to ensure that the safety analyses on the "UMAX" docket (72-1040) remain aligned with the material referenced in the HI-STORM FW FSAR.

The applicable design criteria for the MPC and the HI-TRAC transfer cask are based on their licensing bases in the HI-STORM FW docket. MPC design criteria information extracted from the HI-STORM FW FSAR is provided in "Arial" font throughout this chapter.

Tables 2.0.1, et.seq. in the HI-STORM FW FSAR contain the ITS designations of the subparts of the MPCs and transfer cask that constitute the components of the HI-STORM UMAX system.

The MPC is engineered for a 60 year design life, while satisfying the requirements of 10CFR72. The adequacy of the MPC to meet the above design life is discussed in Section 3.4 of the HI-STORM FW FSAR. The design characteristics of the MPC are described in Section 1.2 of the HI-STORM FW FSAR.

The HI-TRAC VW transfer cask is engineered for a 60 year design life. The adequacy of the HI-TRAC VW to meet the above design life commitment is discussed in Section 3.4 of HI-STORM FW FSAR. The design characteristics of the HI-TRAC VW cask are presented in Section 1.2 the HI-STORM FW FSAR.

A description of the HI-STORM UMAX VVM is provided in Section 1.2. The applicable loads, affected parts under each loading condition, and the applicable structural acceptance criteria are compiled in this Chapter to provide a complete framework for the required qualifying safety analyses in the rest of the safety analysis report. Information consistent with the regulatory requirements related to shielding, thermal performance, confinement, radiological, and operational considerations is also provided. The licensing drawing of the HI-STORM UMAX VVM in Section 1.5 provides information on the necessary *critical characteristics* that define the HI-STORM UMAX system. The constituents of the HI-STORM UMAX ISFSI fall into two broad categories, namely:

a. VVM components

b. ISFSI structures

The safety analyses address both the VVM components and the ISFSI structures. The VVM components consist of:

a. The Cavity Enclosure Container (CEC)

b. The Divider Shell

c. The Closure Lid

The ISFSI Structures consist of:

a. The Support Foundation Pad (SFP)

b. The ISFSI Pad;

c. The Enclosure Wall (optional)

d. The Subgrade and Under-grade

Figure 1.2.1 denotes the key constituent parts of the HI-STORM UMAX VVM and Figure 2.4.4 depicts the subgrade and under-grade nomenclature for the ISFSI. The analysis basis density and shear wave velocities of sub-grade and under-grade spaces are given in Table 2.3.2. The spaces shown in Figure 2.4.4 are further explained below.

a. Space A is the Self-hardening Engineered Subgrade (SES) below the ISFSI Pad after the construction of the SFP. The candidate materials of SES include Controlled Low-Strength Material (CLSM) and lean concrete.

b. Space B is the lateral subgrade that extends by the amount W around the ISFSI where W is the characteristic dimension of the ISFSI. Space A and Space B may be separated by an Enclosure Wall.

c. Space C is the under-grade below the SFP and extending 100 feet down from the top of grade elevation.

d. Space D is the under-grade surrounding Space C extending 100 feet down from the top of grade elevation.

2.0.2 Structural

MPC Design Criteria

The MPC is classified as important-to-safety. The MPC structural components include the fuel basket and the enclosure vessel. The fuel basket is designed and fabricated to meet a more stringent displacement limit under mechanical loadings than those implicit in the stress limits of the ASME code (see Section 2.2 of the HI-STORM FW FSAR). The MPC enclosure vessel is designed and fabricated as a Class 1 pressure vessel in accordance with Section III, Subsection NB of the ASME Code, with certain necessary alternatives, as discussed in Section 2.2 of the HI-STORM FW FSAR. The principal exception to the above Code pertains to the MPC lid, vent and drain port cover plates, and closure ring welds to the MPC lid and shell, as discussed in Section 2.2 of HI-STORM FW FSAR. In addition, Threaded Anchor Locations (TALs) in the MPC lid are designed in accordance with the requirements of NUREG-0612 for critical lifts to facilitate handling of the loaded MPC.

The MPC closure welds are partial penetration welds that are structurally qualified by analysis in Chapter 3 of the HI-STORM FW FSAR. The MPC lid and closure ring welds are inspected by performing a liquid penetrant examination in accordance with the drawings contained in Section 1.5. The integrity of the MPC lid-to-shell weld is further verified by performing a progressive liquid penetrant examination of the weld layers, and a Code pressure test.

The structural analysis of the MPC, in conjunction with the redundant closures and nondestructive examination, pressure testing, and helium leak testing provides assurance of canister closure integrity in lieu of the specific weld joint configuration requirements of Section III, Subsection NB.

Compliance with the ASME Code, with respect to the design and fabrication of the MPC, and the associated justification are discussed in Section 2.2 of the HI-STORM FW FSAR. The MPC design is analyzed for all design basis normal, off-normal, and postulated accident conditions, as defined in Section 2.2 of the HI-STORM FW FSAR.

The required characteristics of the fuel assemblies to be stored in the MPC are limited in accordance with Section 2.1 of the HI-STORM FW FSAR.

HI-TRAC Design Criteria

The HI-TRAC transfer cask includes both structural and non-structural radiation shielding components that are classified as important-to-safety. The structural steel components of the HI-TRAC are designed to meet the stress limits of Section III, Subsection NF, of the ASME Code for normal and off-normal storage conditions. The threaded anchor locations for lifting and handling of the transfer cask are designed in accordance with the requirements of NUREG-0612 and Regulatory Guide 3.61 for interfacing lift points.

The HI-TRAC transfer cask design is analyzed for all normal, off-normal, and design basis accident condition loadings, as defined in Section 2.2 of the HI-STORM FW FSAR. Under accident conditions, the HI-TRAC transfer cask must protect the MPC from unacceptable deformation, provide continued shielding, and remain in a condition such that the MPC can be removed from it. The loads applicable to the HI-TRAC transfer cask are defined in Tables 2.2.6, 2.2.13 and Table 3.1.1 of the HI-STORM FW FSAR. The physical characteristics of each MPC for which the HI-TRAC VW is designed are presented in Subsection 1.2 of the HI-STORM FW FSAR.

HI-STORM VVM Design Criteria

All required information on the design bases and criteria for the VVM is compiled in this Chapter and fulfills the requirements of 10CFR72.24(c) (3) and 72.44(d). The VVM structure described in Chapter 1 is designed for all applicable normal, off-normal, extreme environmental phenomena, and accident condition loadings pursuant to 10CFR72.24(c), 72.122(b) and 72.122(c).

ISFSI Structure Design Criteria

The SFP and the subgrade under the ISFSI pad are categorized as important-to-safety (ITS) structures and are included in the structural analyses in Chapter 3, and in other chapters, as applicable. ACI-318 (05) [2.6.2] is specified as the reference code for the design qualification of the SFP and the ISFSI pad using the load combinations specified in Table 2.4.3. The seismic qualification of the storage system is performed in Chapter 3.

As discussed in sub-section 1.2.2, the subgrade in Space A in Figure 2.4.4 can be made from lean concrete or CLSM. The required structural properties of the subgrade are provided in Tables 2.3.2 and 3.3.4. Chapter 8 provides additional (non-structural) data on the CLSM to ensure the establishment of a durable structure built to meet the service life expectations for the ISFSI.

MATERIAL TYPES

The material types used in the VVM are identified in Table 2.6.2. Material designations used by ASTM and ASME for various product forms are, however, subject to change as these material certifying organizations publish periodic updates of their standards. Material designations adopted by the International Standards Organization (ISO) also affect the type of steels and steel alloys available from suppliers around the world. Therefore, it is necessary to provide for the ability to substitute materials with equivalent materials in the manufacture of the equipment.

The term Equivalent Material* has a specific meaning in this FSAR (and other FSARs in Holtec dockets listed in Table 1.2.1). Equivalent materials are those that can be substituted for each other without adversely affecting the safety function of the SSC (system, structure, and component) in which the substitution is made.

The equivalence of materials is directly tied to the notion of *critical characteristics*. A critical characteristic of a material is a property whose value must be specified and controlled to ensure an SSC will render its intended function. The numerical value of the critical characteristic invariably enters in the safety evaluation of an SSC and therefore its range must be guaranteed. To ensure that the safety calculation is not adversely affected, material properties such as Yield Strength, Ultimate Strength and Elongation must be specified as *minimum* guaranteed values in VVM Components. However, there are certain properties where both minimum and maximum acceptable values are required. In this category lies specific gravity and thermal expansion coefficient for the VVM components.

Table 2.6.3 lists the array of material properties typically required in safety evaluation of an SSC in dry storage and transport applications. The required value of each applicable property, guided by the safety evaluation defines the critical characteristics of the material. The subset of applicable properties for a material depends on the role played by the material. The role of a material in the SSC is divided into three category types, namely structural, thermal, and radiation compliance. The material properties listed in Table 2.6.4 are the ones that apply to the VVM components.

To summarize, the following procedure shall be used to establish acceptable equivalent materials for a particular application.

Criterion i: Functional Adequacy:

Evaluate the guaranteed critical characteristics of the equivalent material against the values required to be used in safety evaluations. The required values of each critical characteristic must be met by the minimum (or maximum) guaranteed values (MGVs of the selected material).

Criterion ii: Chemical and Environmental Compliance:

* This text matter on Equivalent Materials is adapted from previously approved FSARs [2.0.1, 2.6.3].

Perform the necessary evaluations and analyses to ensure the candidate material will not excessively corrode or otherwise degrade in the operating environment.

A material from another designation regime that meets Criteria (i) and (ii) above is deemed to be an acceptable material, and hence, equivalent to the candidate material.

For ITS materials, recourse to equivalent materials shall be made only in the extenuating circumstances where the designated material is not readily available.

As can be ascertained from its definition in the glossary, the *critical characteristics* of the material used in a subcomponent depend on its function. The Closure Lid, for example, serves as a shielding device and as a physical barrier to protect the MPC against loadings under all service conditions, including the Extreme Environmental phenomena. Therefore, the critical characteristics of steel used in the lid are its strength (yield and ultimate), ductility, and fracture resistance.

The appropriate critical characteristics for structural components of the VVM, therefore, are:

a. Material yield strength, σ_y

b. Material ultimate strength, σ_u

c. Elongation, ε

d. Charpy impact strength at the lowest service temperature for the part, C_i (unless exempted by other provisions in the governing code)

Thus, the carbon steel specified in the drawing package can be substituted with different steel so long as each of the four above properties in the replacement material is equal to or greater than the minimum values used in the qualifying analyses. The above *critical characteristics* apply to all materials used in the structural parts of the CEC.

In the event that one or more of the *critical characteristics* of the replacement material is slightly lower than the original material, then the use of the §72.48 process is necessary to ensure that all regulatory predicates for the material substitution are fully satisfied.

In addition to the design configuration and materials of construction, the maximum magnitude of Design Basis Earthquake for the HI-STORM UMAX ISFSI is also specified. A non-linear time-history solution procedure implemented on LS-DYNA is used in Chapter 3 to qualify the ISFSI including the storage system. This same non-linear time-history solution procedure must be used to perform safety evaluation under 10CFR72.212 at a host site. Likewise, the loadings from the extreme environmental phenomena, defined in this chapter, are considered in Chapter 3. Site specific loadings that deviate from those analyzed in Chapter 3 are subject to §72.212 safety evaluations in the manner of all HI-STORM models. For sites where the topography of the land results in an inclined or otherwise uneven top surface of the sub-grade girdling the ISFSI pad, the ISFSI owner, at his option, may require a site specific soil-structure interaction analysis to be performed as an input to the site's 10CFR72.212 safety evaluation. The seismic event selected for the SSI analysis may be the Design Basis Earthquake or another (more severe) earthquake deemed to be more appropriate or bounding by the ISFSI owner and the analysis methodology

shall be implemented on LS-DYNA approved in this FSAR. The safety criteria set forth in this FSAR and the applicable Technical Specification shall be met.

To serve their intended functions, the CEC and Closure Lid shall ensure physical protection, biological shielding, and allow the retrieval of the MPC under all conditions of storage (10 CFR 72.122(l)). Because the VVM is an in-ground structure, drops and tip-over of the VVM are not credible events and, therefore, do not warrant analysis. The load cases germane to establishing the structural adequacy of the VVM pursuant to 10 CFR 72.24(c) are compiled in Table 2.4.1. The physical characteristics of the MPC intended for storage in the VVM are presented in Chapter 1.

The design bases and criteria provided in this Chapter are intended to quantify the safety margins in the VVM design with respect to all applicable loadings that follow from the provisions of 10CFR72.24(c)(3), §72.122(b) and §72.122(c).

2.0.3 Thermal

MPC Design Criteria

The thermal design and operation of the MPC in the HI-STORM UMAX System meets the intent of the review guidance contained in ISG-11, Revision 3 [2.4.6]. Specifically, the ISG-11 provisions that are explicitly invoked and satisfied are:

i. The thermal acceptance criteria for all commercial spent fuel (CSF) authorized by the USNRC for operation in a commercial reactor are unified into one set of requirements.

ii. The maximum value of the calculated temperature for all CSF under long-term normal conditions of storage must remain below 400ºC (752ºF). For short-term operations, including canister drying, helium backfill, and on-site cask transport operations, the fuel cladding temperature must not exceed 400ºC (752ºF) for high burnup fuel (HBF) and 570ºC (1058ºF) for moderate burnup fuel.

iii. The maximum fuel cladding temperature as a result of an off-normal or accident event must not exceed 570ºC (1058ºF).

iv. For HBF, operating restrictions are imposed to limit the maximum temperature excursion during short-term operations to 65ºC (117ºF) and the number of excursions to less than 10.

To achieve compliance with the above criteria, certain design and operational changes are necessary, as summarized below.

i The peak fuel cladding temperature limit (PCT) for long term storage operations and short term operations is generally set at 400ºC (752ºF). However, for MPCs containing all moderate burnup fuel, the fuel cladding temperature limit for short-term operations is set at 570ºC (1058ºF) because the nominal fuel cladding stress is shown to be less than 90 MPa [2.0.2]. Appropriate analyses have been

performed as discussed in Chapter 4 and operating restrictions have been added to ensure these limits are met.

ii. A method of drying, such as forced helium dehydration (FHD) is used if the above temperature limits for short-term operations cannot be met.

iii. The off-normal and accident condition PCT limit remains unchanged at 570ºC (1058ºF).

The MPC cavity is dried, either with FHD or vacuum drying, and then it is backfilled with high purity helium to promote heat transfer and prevent cladding degradation.

The normal condition design temperatures for the stainless steel components in the MPC are provided in Table 2.3.7.

Each MPC model allows for regionalized storage where every storage location is identified by a unique number as shown in Figures 2.1.7 and 2.1.8. The permissible storage cell heat load limits and total design heat load limits are presented in Tables 2.1.8 and 2.1.9 for MPC-37 and MPC-89, respectively. The decay heat loads in Tables 2.1.8 and 2.1.9 are a result of CoC restrictions limiting the permissible aggregate heat load† to 80% of the design basis heat load limits. This limit on the aggregate decay heat will compensate for the uncertainties associated with the modeling and application errors, which will be determined by validating the analysis methods. Due to lack of experimental data to validate the thermal analysis, a thermal test described in Section 10.3 will be performed which will be used to benchmark the CFD thermal model developed in Chapter 4. Specific requirements, such as approved locations for DFCs and non-fuel hardware are given in Section 2.1.

HI-TRAC Design Criteria

The allowable temperatures for the HI-TRAC transfer cask structural steel components are based on the maximum temperature for material properties and allowable stress values provided in Section II of the ASME Code. The allowable temperatures for the structural steel and shielding components of the HI-TRAC are provided in Table 2.3.7. The HI-TRAC is designed for off-normal environmental cold conditions, as discussed in Subsection 2.2.2 of the HI-STORM FW FSAR. The evaluation of the potential for brittle fracture in structural steel materials is presented in Section 3.1 of the HI-STORM FW FSAR.

The HI-TRAC is designed and evaluated for the maximum heat load analyzed for storage operations. The maximum allowable temperature of water in the HI-TRAC jacket is a function of the internal pressure. To preclude over- pressurization of the water jacket due to boiling of the neutron shield liquid (water), the maximum temperature of the water is restricted to be less than the saturation temperature at the shell design pressure. Even though the analysis shows that the water jacket will not over-pressurize, a relief device is placed at the top of the water jacket shell. In addition, the water is precluded from freezing during off-normal cold conditions by limiting the

† The aggregate heat load is defined as the sum of heat loads of all stored fuel assemblies.

minimum allowable operating temperature and by adding ethylene glycol. The thermal characteristics of the fuel for each MPC for which the transfer cask is designed are defined in Section 2.1. The working area ambient temperature limit for loading operations is limited in accordance with Table 2.3.6.

HI-STORM VVM Design Criteria

The HI-STORM VVM rejects heat from the stored MPCs by delivering cool ambient air to the annular space around the MPC. The ambient air undergoes progressive heating and reduction in density as it rises in the cylindrical space surrounding the MPC through convective heat transfer with the MPC shell, and exits the cell through the vertical flue mounted on the central region of the closure lid. The storage cavities have a constant out flow of air which will tend to retard the deposition of air borne particulates and debris in the storage space. The accumulated solids can be vacuumed out of the storage cavity by standard means. As shown in Chapter 4, the VVMs are designed to reject the maximum allowable heat load as defined below in a reliable and testable manner consistent with its important-to-safety designation (10CFR72.128(a)(4)). The VVM is designed for extreme cold conditions. The safety of structural steel material used for the VVM from brittle fracture is discussed in Chapter 8.

2.0.4 Shielding

The HI-TRAC transfer cask provides shielding to maintain occupational exposures ALARA in accordance with 10CFR20, while also maintaining the maximum load on the plant's crane hook below the rated capacity of the crane. As discussed in Subsection 1.2.1 of the HI-STORM FW FSAR, the shielding in HI-TRAC is maximized within the constraint of the allowable weight at a plant site. The HI-TRAC calculated dose rates for a set of reference conditions are reported in Section 5.1 of HI-STORM FW FSAR. These dose rates are used to perform a generic occupational exposure estimate for MPC loading, closure, and transfer operations, as described in Chapter 11 of the HI-STORM FW FSAR. A postulated HI-TRAC accident condition, which includes the loss of the liquid neutron shield (water), is also evaluated in Chapter 5 of the HI-STORM FW FSAR.

The annular area between the MPC outer surface and the HI-TRAC inner surface can be isolated to minimize the potential for surface contamination of the MPC by spent fuel pool water during wet loading operations. The HI-TRAC surfaces expected to require decontamination are coated with a suitable coating. The maximum permissible surface contamination for the HI-TRAC is in accordance with plant-specific procedures and ALARA requirements (discussed in Chapter 11 of the HI-STORM FW FSAR).

The off-site dose for normal operating conditions to any real individual beyond the controlled area boundary is limited by 10CFR72.104(a) to a maximum of 25 mrem/year whole body, 75 mrem/year thyroid, and 25 mrem/year for other critical organs, including contributions from all nuclear fuel cycle operations. Since these limits are dependent on plant operations as well as on

site-specific conditions (e.g., the ISFSI design and proximity to the controlled area boundary, and the number and arrangement of loaded storage casks at the ISFSI), the determination and comparison of ISFSI doses to these limits are necessarily site-specific. Dose rates from the HI-STORM UMAX System are provided in Chapter 5. The determination of site-specific ISFSI dose rates at the site boundary and demonstration of compliance with regulatory limits is to be performed by the licensee for the specific VVM array in accordance with 10CFR72.212.

The HI-STORM UMAX VVM is designed to limit the dose rates for all MPCs to ALARA values. The VVM is also designed to maintain occupational exposures ALARA during MPC transfer operations, in accordance with 10CFR20. The underground location of the VVMs significantly reduces the radiation from the ISFSI at the site boundary compared to an aboveground cask. The calculated VVM dose rates, including dose rates during site construction next to an operating ISFSI, are discussed in Chapter 5.

2.0.5 Criticality

The MPC provides criticality control for all design basis normal, off-normal, and postulated accident conditions, as discussed in Section 6.1 of the HI-STORM FW FSAR. The effective neutron multiplication factor is limited to k_{eff} < 0.95 for fresh (unirradiated) fuel with optimum water moderation with due consideration of all biases, uncertainties, and manufacturing tolerances.

Criticality control is maintained by the geometric spacing of the fuel assemblies and the spatially distributed B-10 isotope in the Metamic-HT fuel basket, and for the PWR MPC model, the additional soluble boron in the MPC water. The minimum specified boron concentration in the purchasing specification for Metamic-HT must be met in every lot of the material manufactured. The guaranteed B-10 value in the neutron absorber, assured by the manufacturing process, is further reduced by 10% (90% credit is taken for the Metamic-HT) to accord with NUREG/CR-5661. No credit is taken for fuel burnup or integral poisons such as gadolinia in BWR fuel. The soluble boron concentration requirements (for PWR fuel only) based on the initial enrichment of the fuel assemblies are delineated in Section 2.1 consistent with the criticality analysis described in Chapter 6 of the HI-STORM FW FSAR [2.0.1].

The HI-STORM UMAX VVM does not perform any criticality control function. The MPCs provide criticality control for all design basis normal, off-normal and postulated accident conditions.

2.0.6 Confinement

The MPC provides for confinement of all radioactive materials for all design basis normal, off-normal, and postulated accident conditions. As discussed in Chapter 7 of the HI-STORM FW, MPC design meets the guidance in the Interim Staff Guidance (ISG)-18 so that leakage of radiological matter from the confinement boundary is non-credible. Therefore, no confinement dose analysis is required or performed. The confinement function of the MPC is verified through pressure testing, helium leak testing, and a

rigorous weld examination regimen executed in accordance with the acceptance test program in Chapter 10 of the HI-STORM FW FSAR.

The HI-TRAC transfer cask does not perform any confinement function. The HI-TRAC provides physical protection and radiation shielding of the MPC contents during MPC loading, unloading, and transfer operations.

The VVM does not perform any confinement function. Confinement during storage is provided by the MPC and is addressed above.

2.0.7 Operations

There are no radioactive effluents that result from storage of transfer operations. Effluents generated during MPC fuel loading are handled by the plant's radioactive waste system and procedures.

The cask operations unique to the HI-STORM UMAX (and common with the HI-STORM 100U in the HI-STORM 100 docket) begin with the arrival of the loaded transfer cask at the HI-STORM UMAX ISFSI. The cask transporter is typically used to move the loaded transfer cask to the ISFSI and to transfer the MPC into the VVM cavities. Generic operating procedures for the HI-STORM UMAX System are provided in Chapter 9. Detailed operating procedures will be developed by the licensee using the information provided in Chapter 9 along with the site-specific requirements that comply with the 10CFR50 Technical Specifications for the plan, and applicable Certificate(s) of Compliance (CoC).

The following overarching requirements apply to all SSCs used in the HI-STORM UMAX operations:

a. All threaded parts shall have provisions to prevent rust build up.
b. All rigging components shall be inspected for visible indications of damage or degradation prior to use.
c. All ancillaries used in heavy load handling shall comply with the brittle fracture criteria set down for cask components in this FSAR.

2.0.8 Acceptance Tests and Maintenance

The acceptance criteria and maintenance program to be applied to the MPC are described in Chapter 10 of the HI-STORM FW FSAR. The operational controls and limits to be applied to the MPC are discussed in Chapter 13. Application of these requirements will assure that the MPC is fabricated, operated, and maintained in a manner that satisfies the design criteria defined in this chapter.

The acceptance criteria and maintenance program to be applied to the HI-TRAC Transfer Cask are described in Chapter 10 of the HI-STORM FW FSAR. The operational controls and limits to be applied to the HI-TRAC are contained in Chapter 13. Application of these requirements will assure that the HI-TRAC is fabricated, operated, and maintained in a manner that satisfies the design criteria given in this chapter.

The fabrication acceptance bases and maintenance program to be applied to the VVMs are described in Chapter 10. Application of these requirements will assure that the VVMs are fabricated and maintained in a manner that satisfies all applicable design criteria.

2.0.9 Decommissioning

The MPC is designed to be transportable in a HI-STAR overpack and is not required to be unloaded prior to shipment off-site.

Decommissioning considerations for the HI-STORM UMAX System including HI-TRAC are addressed in Section 2.11.

Table 2.0.1 – MPC-37 Enclosure Vessel (Drawing # 6505)

Item Number*	Part Name	ITS QA Safety Category
1	Shell, Enclosure Vessel	A
2	Plate, Enclosure Vessel Base	A
3	Plate, Enclosure Vessel Lift Lug	C
4	Plate, Enclosure Vessel Upper Lid	A
5	Plate, Enclosure Vessel Lower Lid	B
6	Ring, Enclosure Vessel Closure	A
7	Block, Enclosure Vessel Vent/Drain Upper	B
8	Port, Enclosure Vessel Vent/Drain	C
9	Plug, Enclosure Vessel Vent/Drain	C
10	Block, Enclosure Vessel Lower Drain	C
12	Block, Enclosure Vessel Vent Shielding	C
13	Plate, Enclosure Vessel Vent/Drain Port Cover	A
16	Purge Tool Port Plug	C
21	Shim, Enclosure Vessel Type 1 PWR Fuel Basket	C
22	Shim, Enclosure Vessel Type 2 PWR Fuel Basket	C

*Item Numbers are non-consecutive because they are consistent with Parts List on drawing.

Table 2.0.2 – Assembly, MPC-37 Fuel Basket (Drawing # 6506)		
Item Number	Part Name	ITS QA Safety Category
1	Panel, Type 1 Cell Wall	A
2	Panel, Type 2 Cell Wall	A
3	Panel, Type 3 Cell Wall	A
4	Panel, Type 4 Cell Wall	A
5	Panel, Type 5 Cell Wall	A
6	Panel, Type 6 Cell Wall	A

Table 2.0.3 – Assembly, MPC-89 Fuel Basket (Drawing # 6507)		
Item Number	Part Name	ITS QA Safety Category
1	Panel, Type 1 Cell Wall	A
2	Panel, Type 2 Cell Wall	A
3	Panel, Type 3 Cell Wall	A
4	Panel, Type 4 Cell Wall	A
5	Panel, Type 5 Cell Wall	A
6	Panel, Type 6 Cell Wall	A
7	Panel, Type 7 Cell Wall	A
8	Panel, Type 8 Cell Wall	A

Table 2.0.4 – MPC-89 Enclosure Vessel (Drawing # 6512)		
Item Number*	Part Name	ITS QA Safety Category
1	Shell, Enclosure Vessel	A
2	Plate, Enclosure Vessel Base	A
3	Plate, Enclosure Vessel Lift Lug	C
4	Plate, Enclosure Vessel Upper Lid	A
5	Plate, Enclosure Vessel Lower Lid	B
6	Ring, Enclosure Vessel Closure	A
7	Block, Enclosure Vessel Vent/Drain Upper	B
8	Port, Enclosure Vessel Vent/Drain	C
9	Plug, Enclosure Vessel Vent/Drain	C
10	Block, Enclosure Vessel Lower Drain	C
12	Block, Enclosure Vessel Vent Shielding	C
13	Plate, Enclosure Vessel Vent/Drain Port Cover	A
16	Purge Tool Port Plug	C
21	Shim, Enclosure Vessel Type 1 BWR Fuel Basket	C
22	Shim, Enclosure Vessel Type 2 BWR Fuel Basket	C
23	Shim, Enclosure Vessel Type 3 BWR Fuel Basket	C

*Item Numbers are non-consecutive because they are consistent with Parts List on drawing.

Table 2.0.5 – HI-TRAC VW – MPC-37 (Drawing # 6514)		
Item Number*	Part Name	ITS QA Safety Category
1	Flange, Bottom	B
3	Hex Bolt, 2-4 ½ UNC X 6" LG.	B
4	Shell, Inner	B
5	Shielding, Gamma	B
6	Flange, Top	A
7	Shell, Water Jacket	B
10	Pipe, Bolt Recess	B
11	Cap, Bolt Recess	B
12	Bottom Lid	B
13	Shell, Outer	B
14	Rib, Extended	B
15	Rib, Short	B

*Item Numbers are non-consecutive because they are consistent with Parts List on drawing.

Table 2.0.6 – HI-TRAC VW – MPC-89 (Drawing # 6799)

Item Number*	Part Name	ITS QA Safety Category
1	Flange, Bottom	B
3	Hex Bolt, 2-4 ½ UNC X 6" LG.	B
4	Shell, Inner	B
5	Shielding, Gamma	B
6	Flange, Top	A
7	Shell, Water Jacket	B
10	Pipe, Bolt Recess	B
11	Cap, Bolt Recess	B
12	Bottom Lid	B
13	Shell, Outer	B
14	Rib, Extended	B
15	Rib, Short	B

*Item Numbers are non-consecutive because they are consistent with Parts List on drawing.

2.1 SPENT FUEL TO BE STORED AND SERVICE LIMITS

2.1.1 Determination of the Design Basis Fuel

A central object in the design of the HI-STORM UMAX System is to ensure that all SNF discharged from the U.S. reactors can be stored in the HI-STORM UMAX MPC upon meeting the burn up, cooling time and content conditions requirements set forth in this FSAR. Publications such as references [2.1.1] and [2.1.2] provide a comprehensive description of fuel discharged from U.S. reactors.

The cell openings in the fuel baskets have been sized to accommodate BWR and PWR assemblies. The cavity length of the MPC will be determined for a specific site to accord with the fuel assembly length used at that site, including non-fuel hardware and damaged fuel containers, as applicable.

Table 2.1.1 summarizes the authorized contents for the HI-STORM UMAX System. Tables 2.1.2 and 2.1.3, which are referenced in Table 2.1.1, provide the fuel characteristics of all groups of fuel assembly types determined to be acceptable for storage in the HI-STORM UMAX System. Any fuel assembly that has fuel characteristics within the range of Tables 2.1.2 and 2.1.3 and meets the other limits specified in Table 2.1.1 is acceptable for storage in the HI-STORM UMAX System. The groups of fuel assembly types presented in Tables 2.1.2 and 2.1.3 are defined as “array/classes” as described in further detail in Chapter 6 of HI-STORM FW FSAR. Table 2.1.4 lists the BWR and PWR fuel assembly designs which are found to govern for three qualification criteria, namely reactivity, shielding, and thermal, or that are used as reference assembly design is those analyses. Additional information on the design basis fuel definition is presented in the following subsections.

2.1.2 Undamaged SNF Specifications

Undamaged fuel is defined in the Glossary.

2.1.3 Damaged SNF and Fuel Debris Specifications

Damaged fuel and fuel debris are defined in the Glossary.

Damaged fuel assemblies and fuel debris will be loaded into damaged fuel containers (DFCs) (Figure 2.1.6) that have mesh screens on the top and bottom. The DFC will have a removable lid to allow the fuel assembly to be inserted. In storage, the lid will be latched in place. DFC’s used to move fuel assemblies will be designed for lifting with either the lid installed or with a separate handling lid. DFC’s used to handle fuel and the associated lifting tools will designed in accordance with the requirements of NUREG-0612. The DFC will be fabricated from structural aluminum or stainless steel. The appropriate structural, thermal, shielding, criticality, and confinement evaluations have been performed to account for damaged fuel and fuel debris and are described in their respective chapters that follow. The limiting design characteristics for damaged fuel

assemblies and restrictions on the number and location of damaged fuel containers authorized for loading in each MPC model are provided in this chapter.

2.1.4 Structural Parameters for Design Basis SNF

The main physical parameters of an SNF assembly applicable to the structural evaluation are the fuel assembly length, cross sectional dimensions, and weight. These parameters, which define the mechanical and structural design, are specified in Subsection 2.1.8. An appropriate axial clearance is provided to prevent interference due to the irradiation and thermal growth of the fuel assemblies.

2.1.5 Thermal Parameters for Design Basis SNF

The principal thermal design parameter for the stored fuel is the fuel's peak cladding temperature (PCT) which is a function of the maximum decay heat per assembly and the decay heat removal capabilities of the HI-STORM UMAX System.

To ensure the permissible PCT limits are not exceeded, Subsection 2.1.9 specifies the maximum allowable decay heat per assembly for each MPC model.

The fuel cladding temperature is also affected by the heat transfer characteristics of the fuel assemblies. The design basis fuel assembly for thermal calculations for both PWR and BWR fuel is provided in Table 2.1.4.

Finally, the axial variation in the heat generation rate in the design basis fuel assembly is defined based on the axial burnup distribution. For this purpose, the data provided in references [2.1.3] and [2.1.4] are utilized and summarized in Table 2.1.5 and Figures 2.1.3 and 2.1.4. These distributions are representative of fuel assemblies with the design basis burnup levels considered. These distributions are used for analyses only, and do not provide a criteria for fuel assembly acceptability for storage in the HI-STORM UMAX System.

2.1.6 Radiological Parameters for Design Basis SNF

The principal radiological design criteria for the HI-STORM UMAX System are the 10CFR72 §104 and §106 operator-controlled boundary dose rate limits, and the requirement to maintain operational dose rates as low as reasonably achievable (ALARA). The radiation dose is directly affected by the gamma and neutron source terms of the assembly, which is a function of the assembly type, and the burnup, enrichment and cooling time of the assemblies. Dose rates are further directly affected by the size and arrangement of the ISFSI, and the specifics of the loading operations. All these parameters are site-dependent, and the compliance with the regulatory dose rate requirements are performed in site-specific calculations. The evaluations here are therefore performed with reference fuel assemblies, and with parameters that result in reasonably conservative dose rates. The reference assemblies given in Table 1.0.4 of the HI-STORM FW FSAR are the predominant assemblies used in the industry.

The design basis dose rates can be met by a variety of burnup levels and cooling times. Table 2.1.1 provides the acceptable ranges of burnup, enrichment and cooling time for all of the authorized fuel assembly array/classes. Table 2.1.5 and Figures 2.1.3 and 2.1.4 provide the axial distribution for the radiological source terms for PWR and BWR fuel assemblies based on the axial burnup distribution. The axial burnup distributions are representative of fuel assemblies with the design basis burnup levels considered. These distributions are used for analyses only, and do not provide a criteria for fuel assembly acceptability for storage in the HI-STORM UMAX System.

Non-fuel hardware, as defined in the Glossary, has been evaluated and is also authorized for storage in the PWR MPCs as specified in Table 2.1.1.

2.1.7 Criticality Parameters for Design Basis SNF

The criticality analyses for the MPC-37 are performed with credit taken for soluble boron in the MPC water during wet loading and unloading operations. Table 2.1.6 provides the required soluble boron concentrations for this MPC.

2.1.8 Summary of Authorized Contents

Tables 2.1.1 through 2.1.3 specify the limits for spent fuel and non-fuel hardware authorized for storage in the HI-STORM FW System. The limits in these tables are derived from the safety analyses described in the following chapters of this FSAR.

2.1.9 Permissible Heat Load for MPC-37 and MPC-89

MPC-89 (BWR) and MPC-37 (PWR) canisters are previously licensed in Docket 72-1032 for storage of spent fuel and are permitted for storage in HI-STORM UMAX with permissible heat loads as specified in Table 2.1.7. As shown in Figures 2.1.7 and 2.1.8 for MPC-37 and MPC-89 respectively, each storage location is associated with a unique cell identification number. The permissible heat loads for each cell in the canister for storage in the HI-STORM UMAX VVM are given in Figure 2.1.19 and Figures 2.1.12 through 2.1.18 for MPC-89 and MPC-37 respectively. The permissible aggregate heat load for storage in MPC-37 and MPC-89 are provided in Tables 2.1.8 and 2.1.9 respectively.

2.1.10 Permissible Heat Load for MPC-24, MPC-32 and MPC-68

The authorized heat loads in the HI-STORM 100 docket for the MPCs certified for storage in the HI-STORM 100 will be used to determine the acceptability of storing them in HI-STORM UMAX. These analyses will be performed to characterize the thermal behavior of the "UMAX" system; they are not intended to secure certification of the MPCs in docket # 72-1014 at this time.

Regionalized loading of SNF in two regions are permitted in MPC-24, MPC-32 and MPC-68 models. The definition of the two regions for each MPC model is provided in Table 2.1.10. The inner region (Region 1) and the outer region (Region 2), shown in Figures 2.1.9, 2.1.10 and

2.1.11 for different MPC types have maximum permitted specific heat loads denoted by q_1 and q_2, respectively. The maximum permitted values of q_1 and q_2 are related through the ratio X, where,

$X = q_1/q_2$.

The special case where q_1 and q_2 are equal (X= 1) is referred to as “uniform storage”. Table 2.1.11 provides the maximum allowable decay heat per fuel storage location for ZR-clad fuel in a uniform fuel loading pattern for each MPC. (Regionalized fuel loading is not permitted in MPC-68F.) The following procedure is specified in Docket # 72-1014 to determine q_1 and q_2 for any chosen value of X:

(i) Choose a value of X in the permissible range ($0.5 \le X \le 3$)

(ii) Calculate q_2 using the following equation:

$$q_2 = \frac{2 \times Q_d}{\left(1 + X^y\right) \times \left(n_1 \times X + n_2\right)}$$

where:

$y = 0.23/X^{0.1}$
q_2 = Maximum allowable decay heat per fuel storage location in Region 2 (kW)
Q_d= Maximum uniform storage MPC decay heat (34 kW)
X = Ratio of q_1 to q_2 chosen in Step (i)
n_1 = Number of fuel storage locations in Region 1 from Table 2.1.10
n_2 = Number of fuel storage locations in Region 2 from Table 2.1.10

(iii) Calculate q_1 using the following equation:

$$q_1 = X \times q_2$$

Additional details of maximum allowable heat load per fuel storage location are provided in Section 2.1.9.1 of the HI-STORM 100 FSAR [2.6.3].

Table 2.1.1*

MATERIAL TO BE STORED

PARAMETER	VALUE (Note 1)	
	MPC-37	MPC-89
Fuel Type	Uranium oxide undamaged fuel assemblies, damaged fuel assemblies, and fuel debris meeting the limits in Table 2.1.2 for the applicable array/class.	Uranium oxide undamaged fuel assemblies, damaged fuel assemblies, with or without channels, fuel debris meeting the limits in Table 2.1.3 for the applicable array/class.
Cladding Type	ZR (see Glossary for definition)	ZR (see Glossary for definition)
Maximum Initial Rod Enrichment	Depending on soluble boron levels and assembly array/class as specified in Table 2.1.6	≤ 5.0 wt. % U-235
Post-irradiation cooling time and average burnup per assembly	Minimum Cooling Time: 3 years Maximum Assembly Average Burnup: 68.2 GWd/mtU	Minimum Cooling Time: 3 years Maximum Assembly Average Burnup: 65 GWd/mtU

*** The text matter in the "arial" font is excerpted from the HI-STORM FW FSAR with minor editorial changes ,as appropriate**

Table 2.1.1*

MATERIAL TO BE STORED

PARAMETER	**VALUE (Note 1)**	
	MPC-37	MPC-89
Non-fuel hardware post-irradiation cooling time and burnup	Minimum Cooling Time: 3 years Maximum Burnup†: - BPRAs, WABAs and vibration suppressors: 60 GWd/mtU - TPDs, NSAs, APSRs, RCCAs, CRAs, CEAs, water displacement guide tube plugs and orifice rod assemblies: 630 GWd/mtU - ITTRs: not applicable	N/A
Decay heat per fuel storage location	Regionalized Loading: See Table 2.1.8	Regionalized Loading: See Table 2.1.9
Fuel Assembly Nominal Length (in.)	Minimum: 157 (with NFH) Reference: 167.2 (with NFH) Maximum: 199.2 (with NFH and DFC)	Minimum: 171 Reference: 176.5 Maximum: 181.5 (with DFC)
Fuel Assembly Width (in.)	≤ 8.54 (nominal design)	≤ 5.95 (nominal design)

† Burnups for non-fuel hardware are to be determined based on the burnup and uranium mass of the fuel assemblies in which the component was inserted during reactor operation. Burnup not applicable for ITTRs since installed post-irradiation.

Table 2.1.1*

MATERIAL TO BE STORED

PARAMETER	**VALUE (Note 1)**	
	MPC-37	MPC-89
Fuel Assembly Weight (lb)	Reference: 1600 (without NFH) 1750 (with NFH), 1850 (with NFH and DFC) Maximum: 2050 (including NFH and DFC) [NFH means Non-Fuel Hardware]	Reference: 750 (without DFC), 850 (with DFC) Maximum: 850 (including DFC) [DFC means Damaged Fuel Canister]

Table 2.1.1*

MATERIAL TO BE STORED

PARAMETER	VALUE (Note 1)	
	MPC-37	MPC-89
Other Limitations	▪ Quantity is limited to 37 undamaged ZR clad PWR fuel assemblies with or without non-fuel hardware. Up to 12 damaged fuel containers containing PWR damaged fuel and/or fuel debris may be stored in the locations denoted in Figure 2.1.1 with the remaining basket cells containing undamaged ZR fuel assemblies, up to a total of 37. ▪ One NSA. ▪ Up to 30 BPRAs. ▪ BPRAs, TPDs, WABAs, water displacement guide tube plugs, orifice rod assemblies, and/or vibration suppressor inserts may be stored with fuel assemblies in any fuel cell location. ▪ CRAs, RCCAs, CEAs, NSAs, and/or APSRs may be stored with fuel assemblies in fuel cell locations specified in Figure 2.1.5.	Quantity is limited to 89 undamaged ZR clad BWR fuel assemblies. Up to 16 damaged fuel containers containing BWR damaged fuel and/or fuel debris may be stored in locations denoted in Figure 2.1.2 with the remaining basket cells containing undamaged ZR fuel assemblies, up to a total of 89.

Table 2.1.2

PWR FUEL ASSEMBLY CHARACTERISTICS (Note 1)

Fuel Assembly Array/ Class	**14x14 A**	**14x14 B**	**14x14 C**	**15x15 B**	**15x15 C**
No. of Fuel Rod Locations	179	179	176	204	204
Fuel Clad O.D. (in.)	≥ 0.400	≥ 0.417	≥ 0.440	≥ 0.420	≥ 0.417
Fuel Clad I.D. (in.)	≤ 0.3514	≤ 0.3734	≤ 0.3880	≤ 0.3736	≤ 0.3640
Fuel Pellet Dia. (in.) (Note 3)	≤ 0.3444	≤ 0.3659	≤ 0.3805	≤ 0.3671	≤ 0.3570
Fuel Rod Pitch (in.)	≤ 0.556	≤ 0.556	≤ 0.580	≤ 0.563	≤ 0.563
Active Fuel Length (in.)	≤ 150	≤ 150	≤ 150	≤ 150	≤ 150
No. of Guide and/or Instrument Tubes	17	17	5 (Note 2)	21	21
Guide/Instrument Tube Thickness (in.)	≥ 0.017	≥ 0.017	≥ 0.038	≥ 0.015	≥ 0.0165

Table 2.1.2 (continued)

PWR FUEL ASSEMBLY CHARACTERISTICS (Note 1)

Fuel Assembly Array/Class	**15x15 D**	**15x15 E**	**15x15 F**	**15x15 H**	**15x15 I**
No. of Fuel Rod Locations	208	208	208	208	216
Fuel Clad O.D. (in.)	≥ 0.430	≥ 0.428	≥ 0.428	≥ 0.414	≥ 0.413
Fuel Clad I.D. (in.)	≤ 0.3800	≤ 0.3790	≤ 0.3820	≤ 0.3700	≤ 0.3670
Fuel Pellet Dia. (in.) (Note 3)	≤ 0.3735	≤ 0.3707	≤ 0.3742	≤ 0.3622	≤ 0.3600
Fuel Rod Pitch (in.)	≤ 0.568	≤ 0.568	≤ 0.568	≤ 0.568	≤ 0.550
Active Fuel Length (in.)	≤ 150	≤ 150	≤ 150	≤ 150	≤ 150
No. of Guide and/or Instrument Tubes	17	17	17	17	9 (Note 4)
Guide/Instrument Tube Thickness (in.)	≥ 0.0150	≥ 0.0140	≥ 0.0140	≥ 0.0140	≥ 0.0140

Table 2.1.2 (continued)

PWR FUEL ASSEMBLY CHARACTERISTICS (Note 1)

Fuel Assembly Array and Class	**16x16 A**	**17x17A**	**17x17 B**	**17x17 C**	**17x17 D**	**17x17 E**
No. of Fuel Rod Locations	236	264	264	264	264	265
Fuel Clad O.D. (in.)	≥ 0.382	≥ 0.360	≥ 0.372	≥ 0.377	≥ 0.372	≥ 0.372
Fuel Clad I.D. (in.)	≤ 0.3350	≤ 0.3150	≤ 0.3310	≤ 0.3330	≤ 0.3310	≤ 0.3310
Fuel Pellet Dia. (in.) (Note 3)	≤ 0.3255	≤ 0.3088	≤ 0.3232	≤ 0.3252	≤ 0.3232	≤ 0.3232
Fuel Rod Pitch (in.)	≤ 0.506	≤ 0.496	≤ 0.496	≤ 0.502	≤ 0.496	≤ 0.496
Active Fuel length (in.)	≤ 150	≤ 150	≤ 150	≤ 150	≤ 170	≤ 170
No. of Guide and/or Instrument Tubes	5 (Note 2)	25	25	25	25	24
Guide/Instrument Tube Thickness (in.)	≥ 0.0350	≥ 0.016	≥ 0.014	≥ 0.020	≥ 0.014	≥ 0.014

Notes:

1. All dimensions are design nominal values. Maximum and minimum dimensions are specified to bound variations in design nominal values among fuel assemblies within a given array/class.
2. Each guide tube replaces four fuel rods.
3. Annular fuel pellets are allowed in the top and bottom 12” of the active fuel length.
4. One Instrument Tube and eight Guide Bars (Solid ZR).

Table 2.1.3

BWR FUEL ASSEMBLY CHARACTERISTICS (Note 1)

Fuel Assembly Array and Class	**7x7 B**	**8x8 B**	**8x8 C**	**8x8 D**	**8x8 E**
Maximum Planar-Average Initial Enrichment (wt.% ^{235}U)	$\leq$ 4.8	$\leq$ 4.8	$\leq$ 4.8	$\leq$ 4.8	$\leq$ 4.8
No. of Fuel Rod Locations	49	63 or 64	62	60 or 61	59
Fuel Clad O.D. (in.)	$\geq$ 0.5630	$\geq$ 0.4840	$\geq$ 0.4830	$\geq$ 0.4830	$\geq$ 0.4930
Fuel Clad I.D. (in.)	$\leq$ 0.4990	$\leq$ 0.4295	$\leq$ 0.4250	$\leq$ 0.4230	$\leq$ 0.4250
Fuel Pellet Dia. (in.)	$\leq$ 0.4910	$\leq$ 0.4195	$\leq$ 0.4160	$\leq$ 0.4140	$\leq$ 0.4160
Fuel Rod Pitch (in.)	$\leq$ 0.738	$\leq$ 0.642	$\leq$ 0.641	$\leq$ 0.640	$\leq$ 0.640
Design Active Fuel Length (in.)	$\leq$ 150	$\leq$ 150	$\leq$ 150	$\leq$ 150	$\leq$ 150
No. of Water Rods (Note 10)	0	1 or 0	2	1 - 4 (Note 6)	5
Water Rod Thickness (in.)	N/A	$\geq$ 0.034	> 0.00	> 0.00	$\geq$ 0.034
Channel Thickness (in.)	$\leq$ 0.120	$\leq$ 0.120	$\leq$ 0.120	$\leq$ 0.120	$\leq$ 0.100

Table 2.1.3 (continued)

BWR FUEL ASSEMBLY CHARACTERISTICS (Note 1)

Fuel Assembly Array and Class	**8x8F**	**9x9 A**	**9x9 B**	**9x9 C**	**9x9 D**
Maximum Planar-Average Initial Enrichment (wt.% ^{235}U)	≤ 4.5 (Note 12)	≤ 4.8	≤ 4.8	≤ 4.8	≤ 4.8
No. of Fuel Rod Locations	64	74/66 (Note 4)	72	80	79
Fuel Clad O.D. (in.)	≥ 0.4576	≥ 0.4400	≥ 0.4330	≥ 0.4230	≥ 0.4240
Fuel Clad I.D. (in.)	≤ 0.3996	≤ 0.3840	≤ 0.3810	≤ 0.3640	≤ 0.3640
Fuel Pellet Dia. (in.)	≤ 0.3913	≤ 0.3760	≤ 0.3740	≤ 0.3565	≤ 0.3565
Fuel Rod Pitch (in.)	≤ 0.609	≤ 0.566	≤ 0.572	≤ 0.572	≤ 0.572
Design Active Fuel Length (in.)	≤ 150	≤ 150	≤ 150	≤ 150	≤ 150
No. of Water Rods (Note 10)	N/A (Note 2)	2	1 (Note 5)	1	2
Water Rod Thickness (in.)	≥ 0.0315	> 0.00	> 0.00	≥ 0.020	≥ 0.0300
Channel Thickness (in.)	≤ 0.055	≤ 0.120	≤ 0.120	≤ 0.100	≤ 0.100

Table 2.1.3 (continued)

BWR FUEL ASSEMBLY CHARACTERISTICS (Note 1)

Fuel Assembly Array and Class	**9x9 E (Note 3)**	**9x9 F (Note 3)**	**9x9 G**	**10x10 A**	**10x10 B**
Maximum Planar-Average Initial Enrichment (wt.% ^{235}U)	≤ 4.5 (Note 12)	≤ 4.5 (Note 12)	≤ 4.8	≤ 4.8	≤ 4.8
No. of Fuel Rod Locations	76	76	72	92/78 (Note 7)	91/83 (Note 8)
Fuel Clad O.D. (in.)	≥ 0.4170	≥ 0.4430	≥ 0.4240	≥ 0.4040	≥ 0.3957
Fuel Clad I.D. (in.)	≤ 0.3640	≤ 0.3860	≤ 0.3640	≤ 0.3520	≤ 0.3480
Fuel Pellet Dia. (in.)	≤ 0.3530	≤ 0.3745	≤ 0.3565	≤ 0.3455	≤ 0.3420
Fuel Rod Pitch (in.)	≤ 0.572	≤ 0.572	≤ 0.572	≤ 0.510	≤ 0.510
Design Active Fuel Length (in.)	≤ 150	≤ 150	≤ 150	≤ 150	≤ 150
No. of Water Rods (Note 10)	5	5	1 (Note 5)	2	1 (Note 5)
Water Rod Thickness (in.)	≥ 0.0120	≥ 0.0120	≥ 0.0320	≥ 0.030	> 0.00
Channel Thickness (in.)	≤ 0.120	≤ 0.120	≤ 0.120	≤ 0.120	≤ 0.120

Table 2.1.3 (continued)

BWR FUEL ASSEMBLY CHARACTERISTICS (Note 1)

Fuel Assembly Array and Class	**10x10 C**	**10x10 F**	**10x10 G**
Maximum Planar-Average Initial Enrichment (wt.% ^{235}U) (Note 14)	≤ 4.8	≤ 4.7 (Note 13)	≤ 4.6 (Note 12)
No. of Fuel Rod Locations	96	92/78 (Note 7)	96/84
Fuel Clad O.D. (in.)	≥ 0.3780	≥ 0.4035	≥ 0.387
Fuel Clad I.D. (in.)	≤ 0.3294	≤ 0.3570	≤ 0.340
Fuel Pellet Dia. (in.)	≤ 0.3224	≤ 0.3500	≤ 0.334
Fuel Rod Pitch (in.)	≤ 0.488	≤ 0.510	≤ 0.512
Design Active Fuel Length (in.)	≤ 150	≤ 150	≤ 150
No. of Water Rods (Note 10)	5 (Note 9)	2	5 (Note 9)
Water Rod Thickness (in.)	≥ 0.031	≥ 0.030	≥ 0.031
Channel Thickness (in.)	≤ 0.055	≤ 0.120	≤ 0.060

Table 2.1.3 (continued)

BWR FUEL ASSEMBLY CHARACTERISTICS

NOTES:

1. All dimensions are design nominal values. Maximum and minimum dimensions are specified to bound variations in design nominal values among fuel assemblies within a given array/class.
2. This assembly is known as "QUAD+." It has four rectangular water cross segments dividing the assembly into four quadrants.
3. For the SPC 9x9-5 fuel assembly, each fuel rod must meet either the 9x9E or the 9x9F set of limits or clad O.D., clad I.D., and pellet diameter.
4. This assembly class contains 74 total rods; 66 full length rods and 8 partial length rods.
5. Square, replacing nine fuel rods.
6. Variable.
7. This assembly contains 92 total fuel rods; 78 full length rods and 14 partial length rods.
8. This assembly class contains 91 total fuel rods; 83 full length rods and 8 partial length rods.
9. One diamond-shaped water rod replacing the four center fuel rods and four rectangular water rods dividing the assembly into four quadrants.
10. These rods may also be sealed at both ends and contain ZR material in lieu of water.
11. Not Used
12. Fuel assemblies classified as damaged fuel assemblies are limited to 4.0 wt.% U-235.
13. Fuel assemblies classified as damaged fuel assemblies are limited to 4.6 wt.% U-235.

Table 2.1.4

DESIGN BASIS FUEL ASSEMBLY FOR EACH DESIGN CRITERION

Criterion	BWR	PWR
Reactivity/Criticality	GE-12/14 10x10 (Array/Class 10x10A)	Westinghouse 17x17 OFA (Array/Class 17x17B)
Shielding	GE-12/14 10x10	Westinghouse 17x17 OFA
Thermal-Hydraulic	GE-12/14 10x10	Westinghouse 17x17 OFA

Table 2.1.5

NORMALIZED DISTRIBUTION BASED ON BURNUP PROFILE

PWR DISTRIBUTION[1]

Interval	Axial Distance From Bottom of Active Fuel (% of Active Fuel Length)	Normalized Distribution
1	0% to 4-1/6%	0.5485
2	4-1/6% to 8-1/3%	0.8477
3	8-1/3% to 16-2/3%	1.0770
4	16-2/3% to 33-1/3%	1.1050
5	33-1/3% to 50%	1.0980
6	50% to 66-2/3%	1.0790
7	66-2/3% to 83-1/3%	1.0501
8	83-1/3% to 91-2/3%	0.9604
9	91-2/3% to 95-5/6%	0.7338
10	95-5/6% to 100%	0.4670

1 Reference 2.1.7

Table 2.1.5 (continued)

NORMALIZED DISTRIBUTION BASED ON BURNUP PROFILE

BWR DISTRIBUTION[2]		
Interval	**Axial Distance From Bottom of Active Fuel (% of Active Fuel Length)**	**Normalized Distribution**
1	0% to 4-1/6%	0.2200
2	4-1/6% to 8-1/3%	0.7600
3	8-1/3% to 16-2/3%	1.0350
4	16-2/3% to 33-1/3%	1.1675
5	33-1/3% to 50%	1.1950
6	50% to 66-2/3%	1.1625
7	66-2/3% to 83-1/3%	1.0725
8	83-1/3% to 91-2/3%	0.8650
9	91-2/3% to 95-5/6%	0.6200
10	95-5/6% to 100%	0.2200

2 Reference 2.1.8

Table 2.1.6

Soluble Boron Requirements for MPC-37 Wet Loading and Unloading Operations

Array/Class	All Undamaged Fuel Assemblies		One or More Damaged Fuel Assemblies and/or Fuel Debris	
	Maximum Initial Enrichment $\leq$ 4.0 wt% ^{235}U (ppmb)	Maximum Initial Enrichment 5.0 wt% ^{235}U (ppmb)	Maximum Initial Enrichment $\leq$ 4.0 wt% ^{235}U (ppmb)	Maximum Initial Enrichment 5.0 wt% ^{235}U (ppmb)
All 14x14 and 16x16A	1,000	1,500	1,300	1,800
All 15x15 and 17x17	1,500	2,000	1,800	2,300

Note:

1. For maximum initial enrichments between 4.0 wt% and 5.0 wt% ^{235}U, the minimum soluble boron concentration may be determined by linear interpolation between the minimum soluble boron concentrations at 4.0 wt% and 5.0 wt% ^{235}U.

Table 2.1.7

ALLOWABLE HEAT LOAD FOR MPCS ANALYZED* FOR STORAGE IN THE HI-STORM UMAX CANISTER STORAGE SYSTEM

MPC Type	Docket Number	Permissible Heat Load
All	72-1014	36.9 kW Note 2
MPC-89	72-1032	See Table 2.1.9
MPC-37	72-1032 Short Fuel: 128 inches $\leq$ L < 144 inches Standard Fuel: 144 inches $\leq$ L < 168 inches Long Fuel: L $\geq$ 168 inches (Note 1)	See Table 2.1.8

Notes:

1. L means "nominal active fuel length". The nominal, unirradiated active fuel length of the PWR fuel assembly is used to designate it as “short”, “standard” and “long”.
2. The maximum allowable heat load for MPCs from Docket 72-1014 is specified in supporting CoC amendments 5 to 9.

*This issue of the FSAR seeks certification of only MPC-89 and MPC-37, termed as “governing canisters” in HI-STORM UMAX. The analysis performed on other MPCs is for reference purposes only.

TABLE 2.1.8

HI-STORM UMAX MPC-37 PERMISSIBLE HEAT LOADS

Fuel Type (see Table 2.1.7 for length data)	Description	Helium Backfill Pressure Option (Notes 2)	Heat Load per Storage Cell (Note 3)	Permissible Aggregate Heat Load (Note 1), kW
Short Fuel	Heat Load Chart 1	1	Figure 2.1.12	33.88
	Heat Load Chart 2	2	Figure 2.1.14	33.70
	Heat Load Chart 3	1	Figure 2.1.16	33.53
Standard Fuel	Heat Load Chart 1	1	Figure 2.1.12	33.88
	Heat Load Chart 2	2	Figure 2.1.14	33.70
	Heat Load Chart 3	1	Figure 2.1.17	35.30
Long Fuel	Heat Load Chart 1	1	Figure 2.1.13	35.76
	Heat Load Chart 2	2	Figure 2.1.15	35.57
	Heat Load Chart 3	1	Figure 2.1.18	37.06
Short Fuel	Sub-Design Heat Load	3	Figure 2.1.19	34.28
	Threshold Heat Load	3	Figure 2.1.21	33.46
Standard Fuel	Sub-Design Heat Load	3	Figure 2.1.19	34.28
	Threshold Heat Load	3	Figure 2.1.21	33.46
Long Fuel	Sub-Design Heat Load	3	Figure 2.1.20	36.19
	Threshold Heat Load	3	Figure 2.1.21	33.46

Note 1: The aggregate heat load is defined as a sum of all stored fuel assemblies. Thermal evaluations in Chapter 4 are performed with maximum per storage cell heat load in all locations. However, the CoC restricts the permissible aggregate heat load to the value specified in this table.
Note 2: The helium backfill range is in Table 4.4.6.
Note 3: Decay heat limits must be met for all contents in a fuel storage location (i.e., fuel and non-fuel hardware, as applicable).

TABLE 2.1.9

HI-STORM UMAX MPC-89 PERMISSIBLE HEAT LOAD

Permissible Heat Load per Storage Cell	Helium Backfill Pressure Option (Note 2)	Permissible Aggregate Heat Load (Note 1), kW
Figure 2.1.22	1	36.32
Figure 2.1.23	2	36.72
Figure 2.1.24	2	34.75

Note 1: The aggregate heat load is defined as a sum of all stored fuel assemblies. Thermal evaluations in Chapter 4 are performed with maximum per storage cell heat load in all locations. However, the CoC restricts the permissible aggregate heat load to the value specified in this table..
Note 2: The helium backfill range is in Table 4.4.6

Table 2.1.10

MPC FUEL STORAGE REGIONS FOR MPCS LICENSED IN DOCKET 72-1014

MPC	Number of Storage Cells		Storage Cell IDs*	
	Inner Region (n_1)	Outer Region (n_2)	Inner Region	Outer Region
MPC-24/24E/24EF	12	12	4, 5 8 through 11 14 through 17 20 and 21	All other locations
MPC-32/32F	12	20	7, 8, 12 through 15, 18 through 21, 25 and 26	All other locations
MPC-68/68FF/68M	32	36	11 through 14, 18 through 23, 27 through 32, 37 through 42, 46 through 51, 55 through 58	All other locations

* See Figures 2.1.9 through 2.1.11 for storage cell numbering.

Table 2.1.11

HI-STORM UMAX MPC (LICENSED IN DOCKET 72-1014) DESIGN HEAT EMISSION RATES FOR UNIFORM LOADING PATTERN

MPC	Decay Heat (kW)	
	Per Intact Fuel Assembly	Per MPC
MPC-24/24E/24EF	1.416	34
MPC-32/32F	1.062	34
MPC-68/68FF/68M	0.5	34

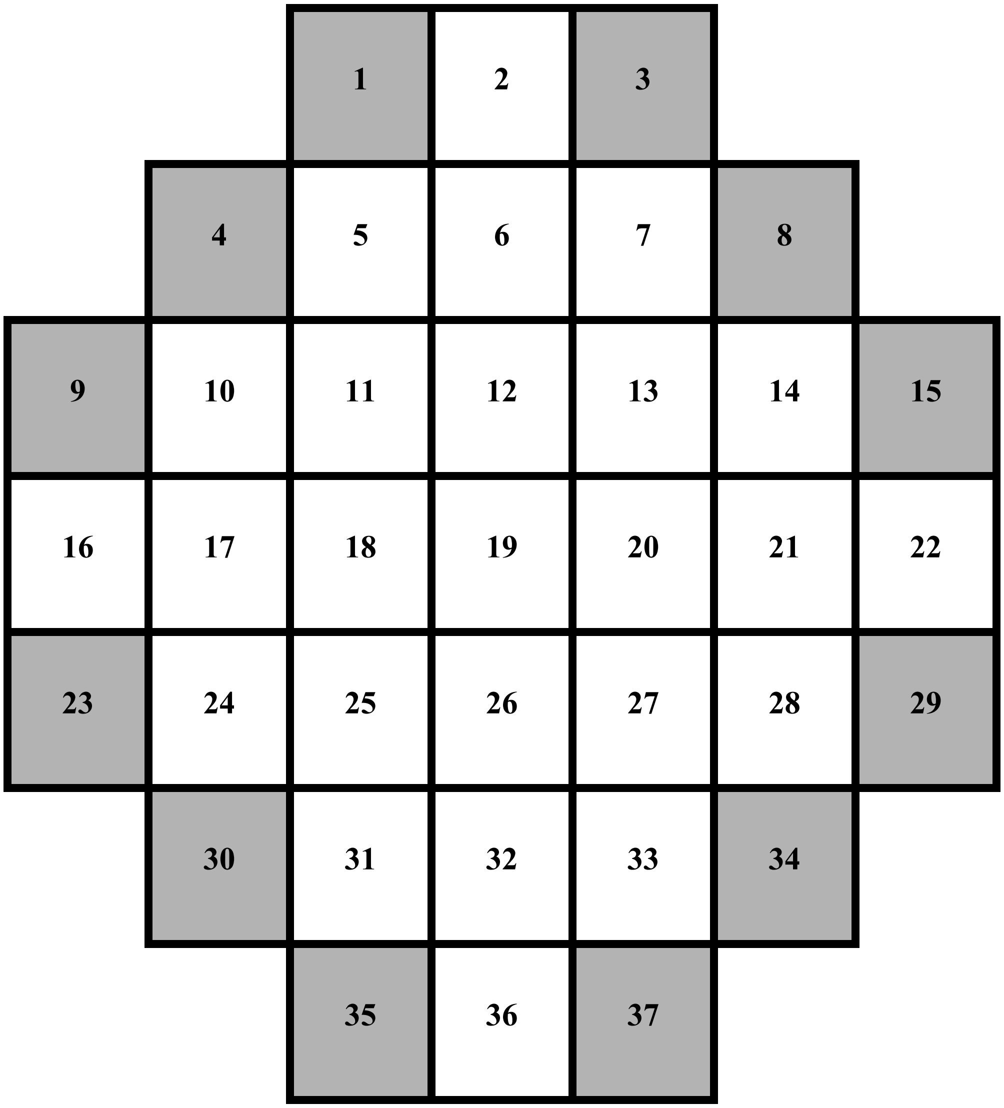


Figure 2.1.1 Location of DFCs for Damaged Fuel or Fuel Debris
in the MPC-37(Shaded Cells)

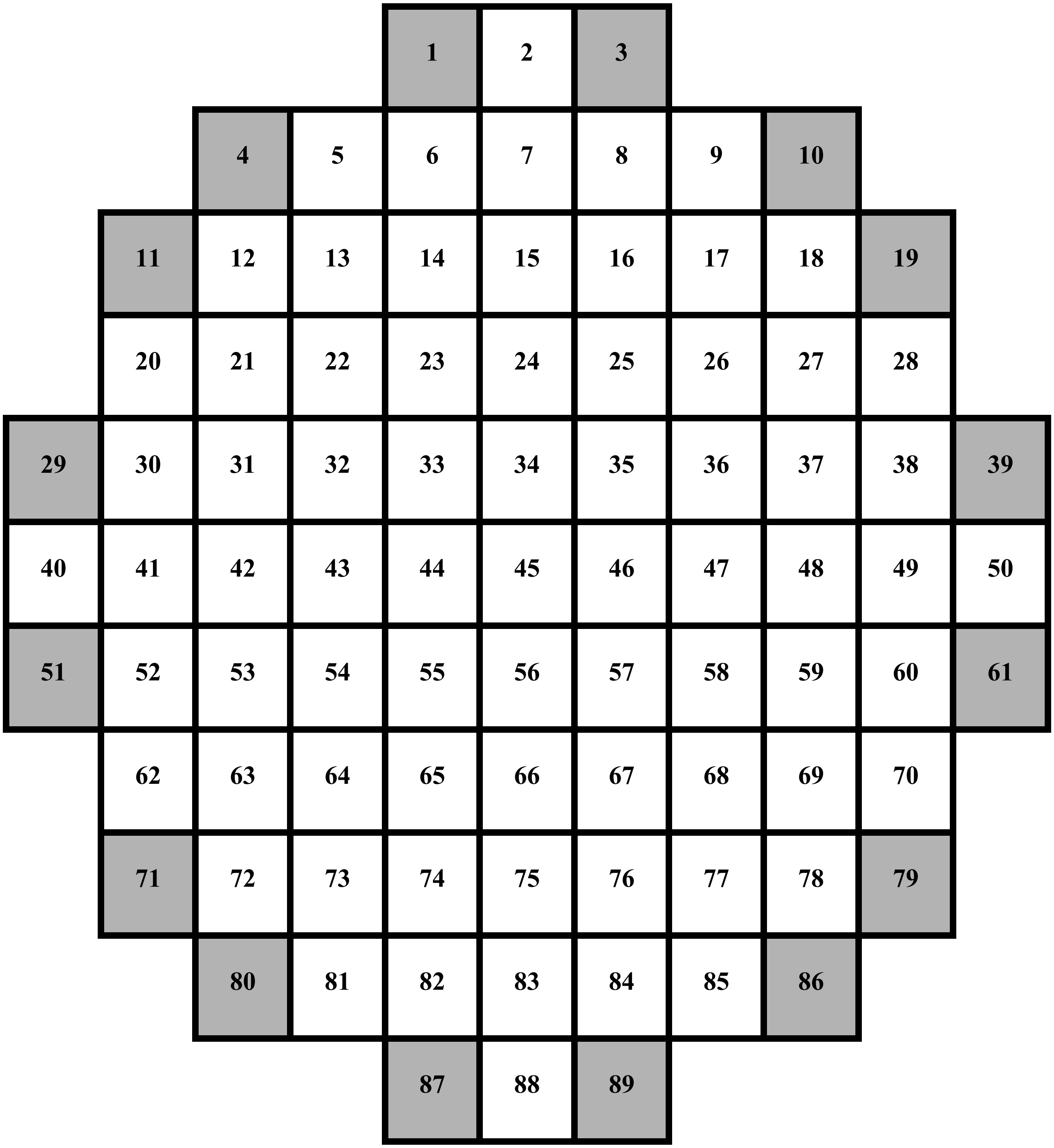


Figure 2.1.2 Location of DFCs for Damaged Fuel or Fuel Debris in the MPC-89 (Shaded Cells)

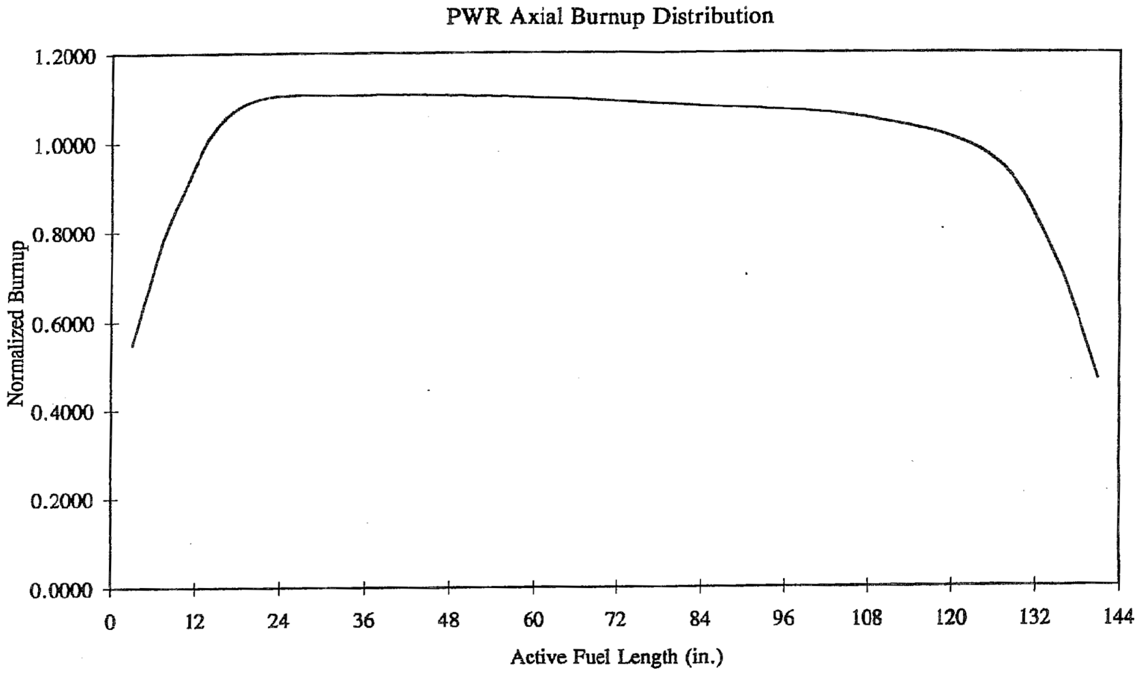


Figure 2.1.3 PWR Axial Burnup Profile with Normalized Distribution

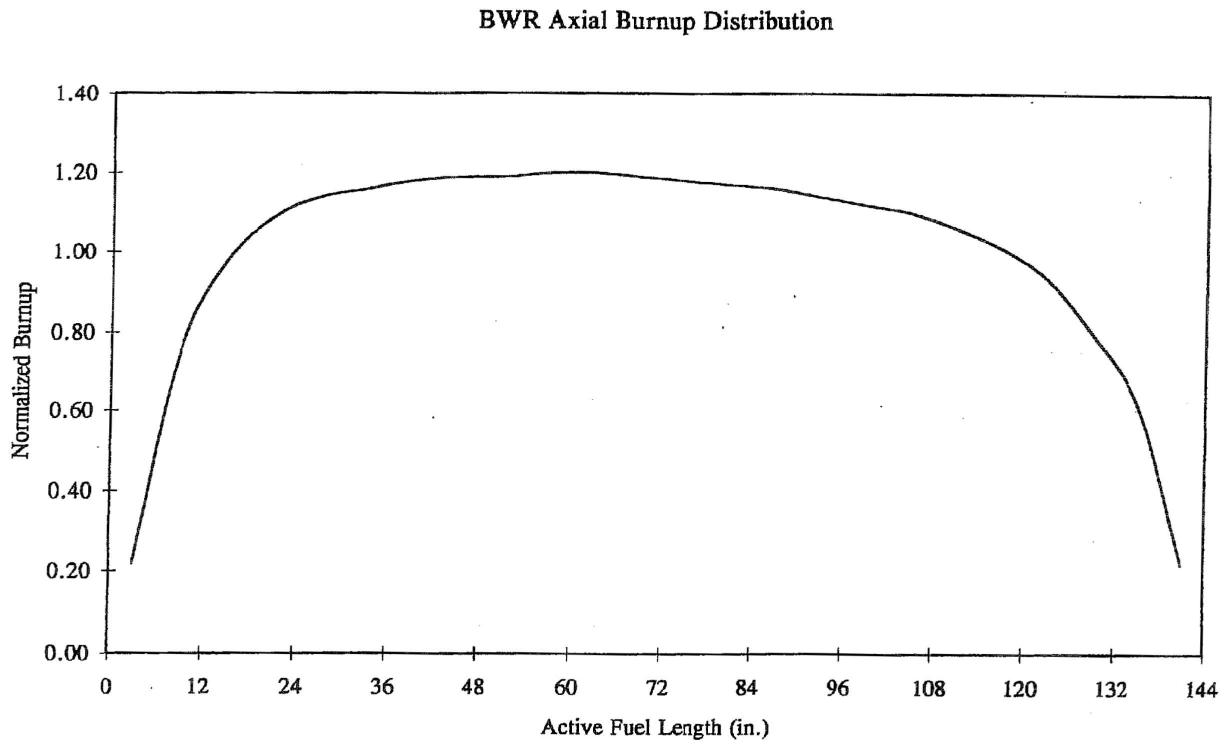


Figure 2.1.4 BWR Axial Burnup Profile with Normalized Distribution

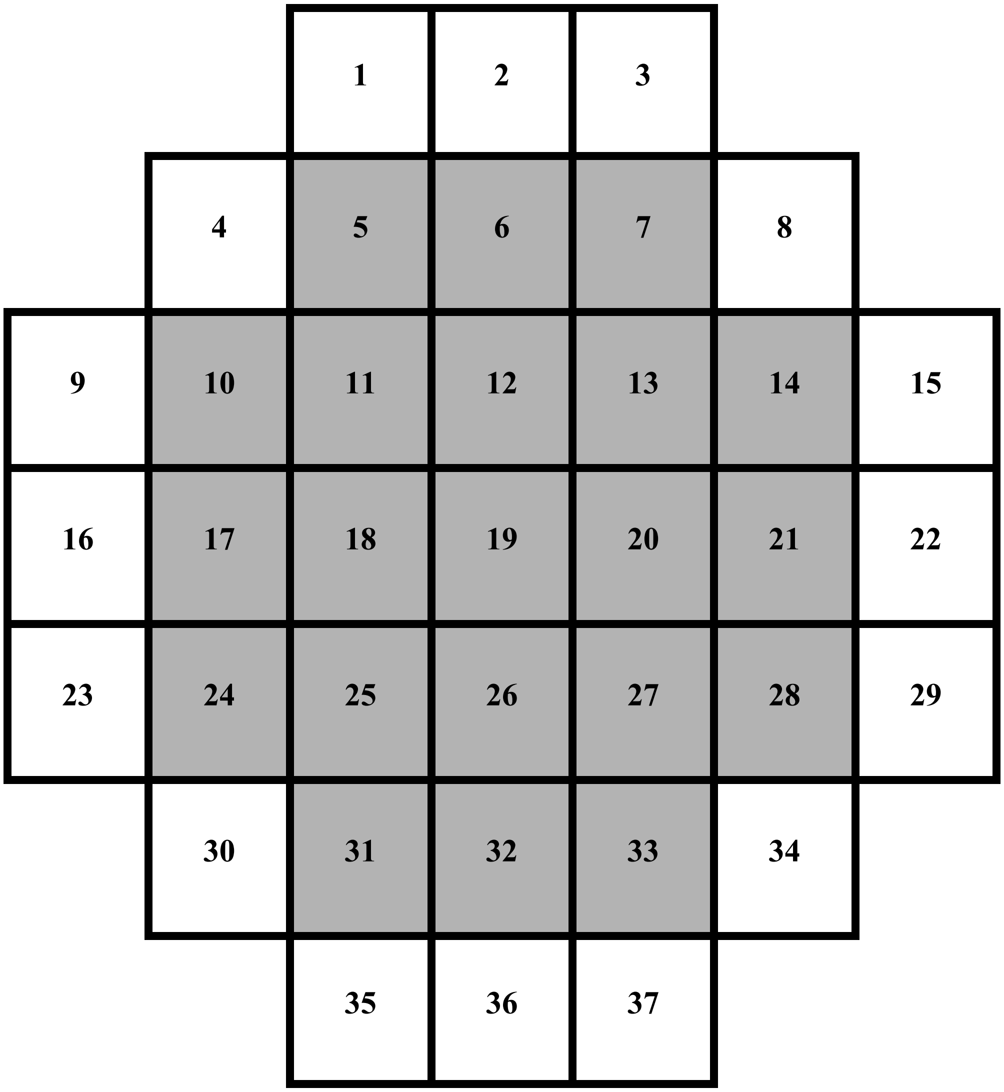


Figure 2.1.5: Location of NSAs, APSRs, RCCAs, CEAs, and CRAs in the MPC-37 (Shaded Cells)

[PROPRIETARY INFORMATION WITHHELD IN ACCORDANCE WITH 10 CFR 2.390]

Figure 2.1.6: Damaged Fuel Container (Typical)

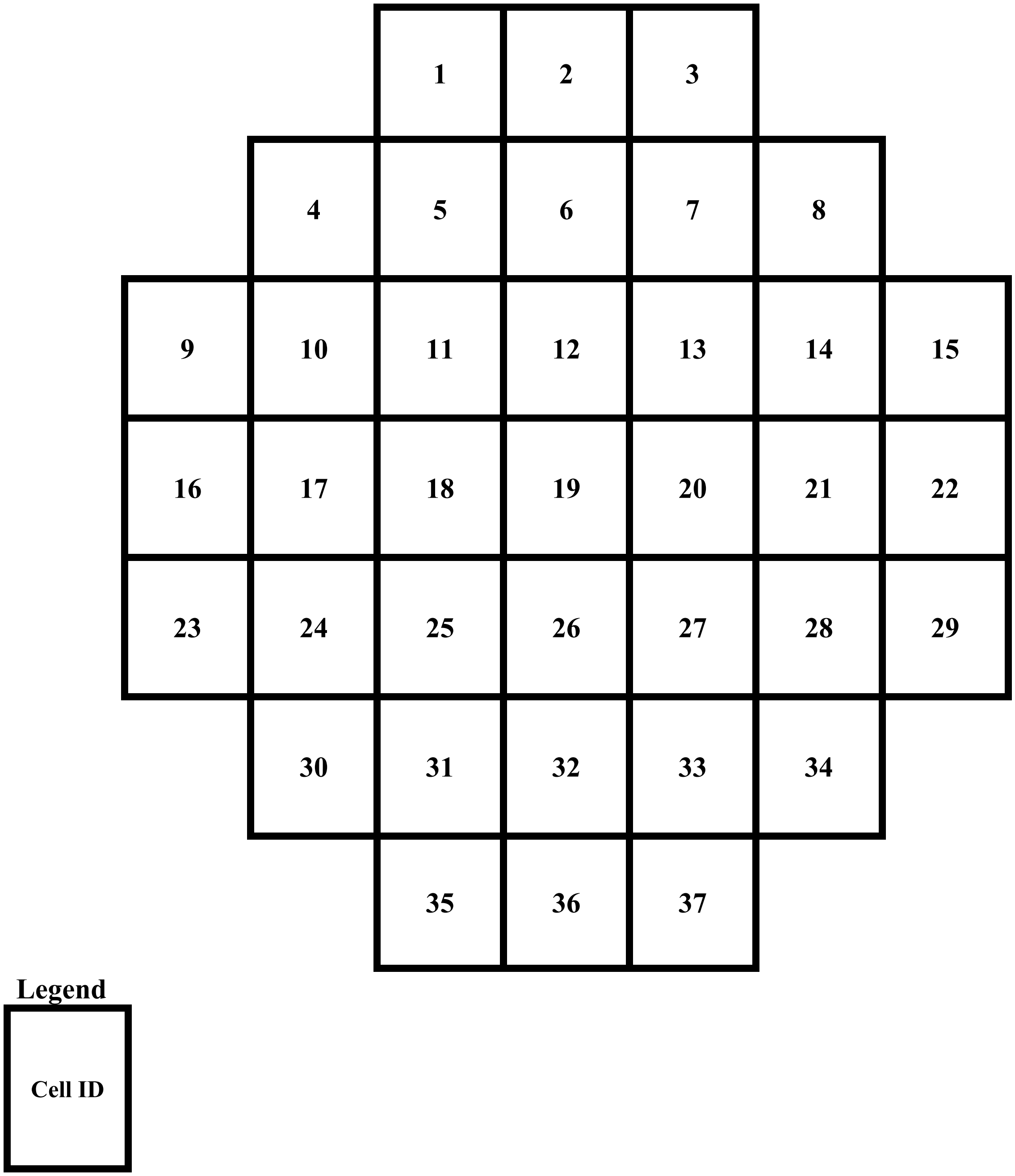


Figure 2.1.7: Cell Identification in MPC-37 Basket

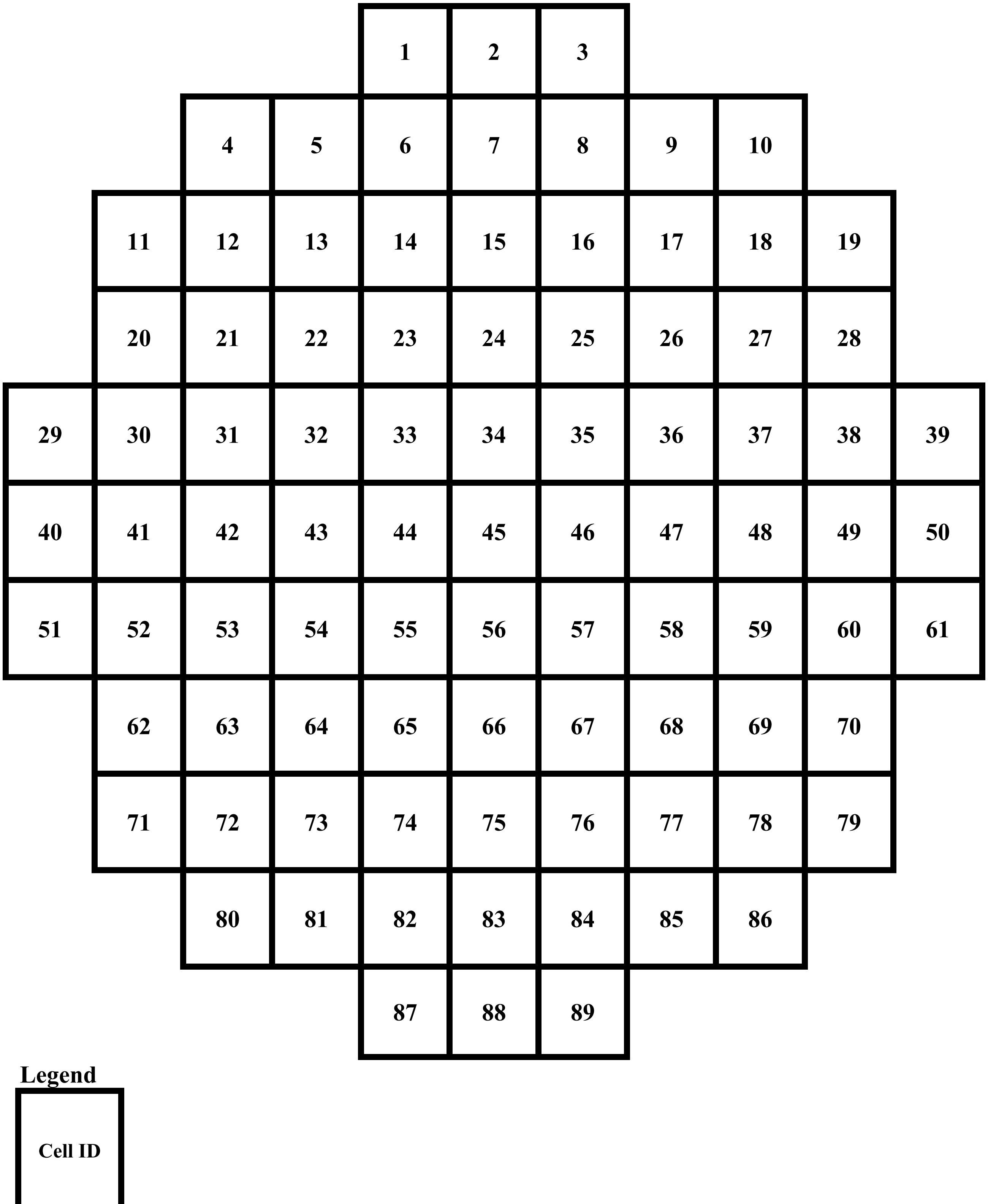


Figure 2.1.8: Cell Identification in MPC-89 Basket

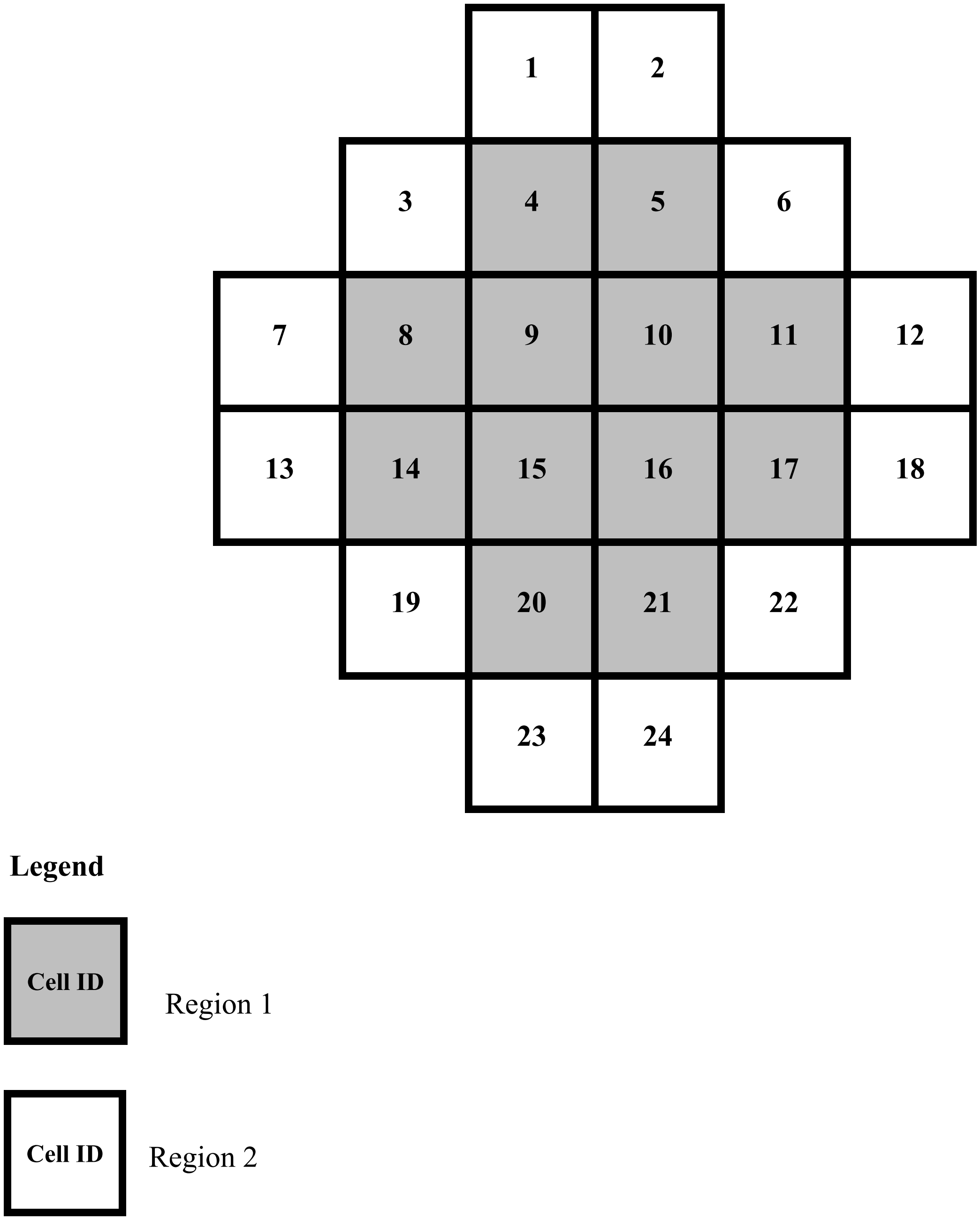


Figure 2.1.9: MPC-24/24E/24EF Cell Identification

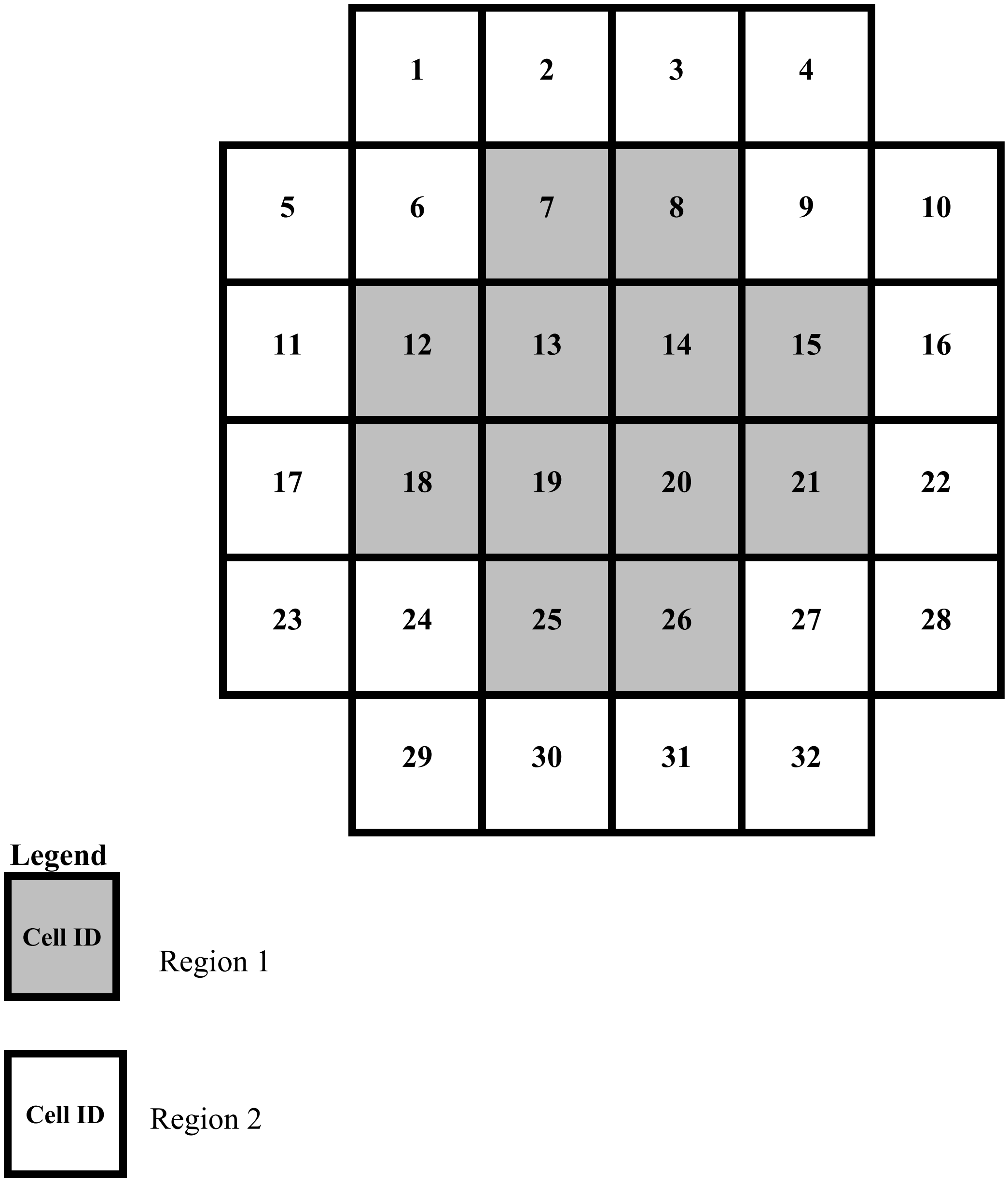


Figure 2.1.10: MPC-32/32F Cell Identification

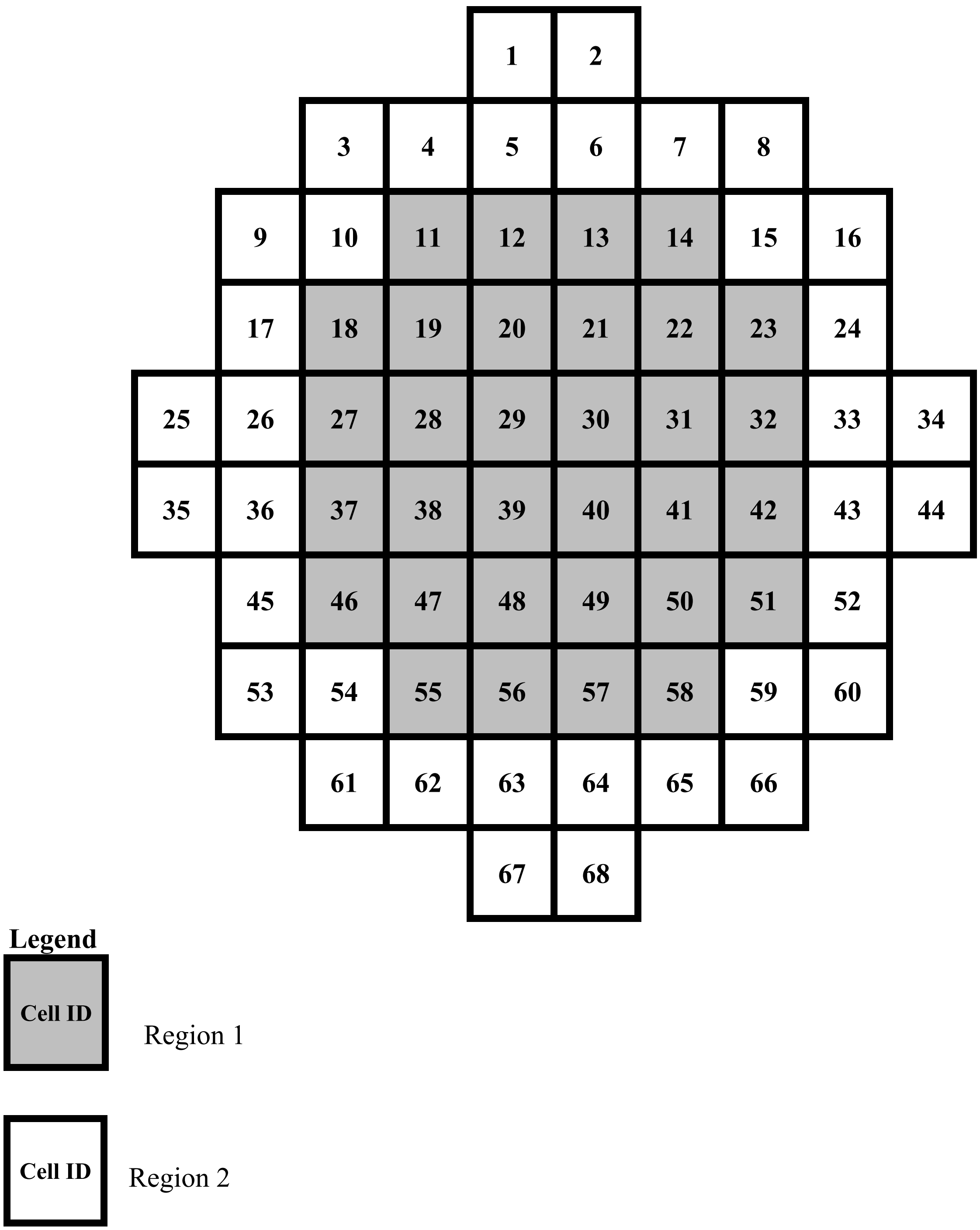


Figure 2.1.11: MPC-68/68F/68FF/68M Cell Identification

		0.873	**0.873**	**0.873**		
	0.873	**1.602**	**1.602**	**1.602**	**0.873**	
0.873	**1.602**	**1.017**	**1.017**	**1.017**	**1.602**	**0.873**
0.873	**1.602**	**1.017**	**1.017**	**1.017**	**1.602**	**0.873**
0.873	**1.602**	**1.017**	**1.017**	**1.017**	**1.602**	**0.873**
	0.873	**1.602**	**1.602**	**1.602**	**0.873**	
		0.873	**0.873**	**0.873**		

Figure 2.1.12: HI-STORM UMAX MPC-37 Heat Load Chart 1 for "Short" and "Standard (Reference)" Fuel

(All storage cell heat loads are in kW)

Note that this figure shows the per cell heat load limit for storage. The total permissible aggregate heat load may be less than the sum of each individual cell heat load. See Table 2.1.8 for corresponding permissible aggregate heat load and the helium backfill option.

		0.922	**0.922**	**0.922**		
	0.922	**1.691**	**1.691**	**1.691**	**0.922**	
0.922	**1.691**	**1.074**	**1.074**	**1.074**	**1.691**	**0.922**
0.922	**1.691**	**1.074**	**1.074**	**1.074**	**1.691**	**0.922**
0.922	**1.691**	**1.074**	**1.074**	**1.074**	**1.691**	**0.922**
	0.922	**1.691**	**1.691**	**1.691**	**0.922**	
		0.922	**0.922**	**0.922**		

Figure 2.1.13: HI-STORM UMAX MPC-37 Heat Load Chart 1 for "Long" Fuel

(All storage cell heat loads are in kW)

Note that this figure shows the per cell heat load limit for storage. The total permissible aggregate heat load may be less than the sum of each individual cell heat load. See Table 2.1.8 for corresponding permissible aggregate heat load and the helium backfill option.

		1.215	1.215	1.215		
	1.215	1.080	1.080	1.080	1.215	
1.215	1.080	1.080	1.080	1.080	1.080	1.215
1.215	1.080	1.080	1.080	1.080	1.080	1.215
1.215	1.080	1.080	1.080	1.080	1.080	1.215
	1.215	1.080	1.080	1.080	1.215	
		1.215	1.215	1.215		

Figure 2.1.14: HI-STORM UMAX MPC-37 Heat Load Chart 2 for "Short" and "Standard (Reference)" Fuel

(All storage cell heat loads are in kW)

Note that this figure shows the per cell heat load limit for storage. The total permissible aggregate heat load may be less than the sum of each individual cell heat load. See Table 2.1.8 for corresponding permissible aggregate heat load and the helium backfill option.

		1.283	**1.283**	**1.283**		
	1.283	**1.140**	**1.140**	**1.140**	**1.283**	
1.283	**1.140**	**1.140**	**1.140**	**1.140**	**1.140**	**1.283**
1.283	**1.140**	**1.140**	**1.140**	**1.140**	**1.140**	**1.283**
1.283	**1.140**	**1.140**	**1.140**	**1.140**	**1.140**	**1.283**
	1.283	**1.140**	**1.140**	**1.140**	**1.283**	
		1.283	**1.283**	**1.283**		

Figure 2.1.15: HI-STORM UMAX MPC-37 Heat Load Chart 2 for “Long” Fuel

(All storage cell heat loads are in kW)

Note that this figure shows the per cell heat load limit for storage. The total permissible aggregate heat load may be less than the sum of each individual cell heat load. See Table 2.1.8 for corresponding permissible aggregate heat load and the helium backfill option.

		0.922	**0.922**	**0.922**		
	0.922	**1.520**	**1.520**	**1.520**	**0.922**	
0.922	**1.710**	**0.950**	**0.950**	**0.950**	**1.710**	**0.922**
0.922	**1.520**	**0.950**	**0.570**	**0.950**	**1.520**	**0.922**
0.922	**1.710**	**0.950**	**0.950**	**0.950**	**1.710**	**0.922**
	0.922	**1.520**	**1.520**	**1.520**	**0.922**	
		0.922	**0.922**	**0.922**		

Figure 2.1.16: HI-STORM UMAX MPC-37 Heat Load Chart 3 for “Short” Fuel

(All storage cell heat loads are in kW)

Note that this figure shows the per cell heat load limit for storage. The total permissible aggregate heat load may be less than the sum of each individual cell heat load. See Table 2.1.8 for corresponding permissible aggregate heat load and the helium backfill option.

		0.970	**0.970**	**0.970**		
	0.970	**1.600**	**1.600**	**1.600**	**0.970**	
0.970	**1.800**	**1.000**	**1.000**	**1.000**	**1.800**	**0.970**
0.970	**1.600**	**1.000**	**0.600**	**1.000**	**1.600**	**0.970**
0.970	**1.800**	**1.000**	**1.000**	**1.000**	**1.800**	**0.970**
	0.970	**1.600**	**1.600**	**1.600**	**0.970**	
		0.970	**0.970**	**0.970**		

Figure 2.1.17: HI-STORM UMAX MPC-37 Heat Load Chart 3 for “Standard (Reference)” Fuel

(All storage cell heat loads are in kW)

Note that this figure shows the per cell heat load limit for storage. The total permissible aggregate heat load may be less than the sum of each individual cell heat load. See Table 2.1.8 for corresponding permissible aggregate heat load and the helium backfill option.

		1.019	**1.019**	**1.019**		
	1.019	**1.680**	**1.680**	**1.680**	**1.019**	
1.019	**1.890**	**1.050**	**1.050**	**1.050**	**1.890**	**1.019**
1.019	**1.680**	**1.050**	**0.630**	**1.050**	**1.680**	**1.019**
1.019	**1.890**	**1.050**	**1.050**	**1.050**	**1.890**	**1.019**
	1.019	**1.680**	**1.680**	**1.680**	**1.019**	
		1.019	**1.019**	**1.019**		

Figure 2.1.18: HI-STORM UMAX MPC-37 Heat Load Chart 3 for "Long" Fuel

(All storage cell heat loads are in kW)

Note that this figure shows the per cell heat load limit for storage. The total permissible aggregate heat load may be less than the sum of each individual cell heat load. See Table 2.1.8 for corresponding permissible aggregate heat load and the helium backfill option.

		0.785	**0.785**	**0.785**		
	0.785	**1.441**	**1.441**	**1.441**	**0.785**	
0.785	**1.441**	**0.915**	**0.915**	**0.915**	**1.441**	**0.785**
0.785	**1.441**	**0.915**	**0.915**	**0.915**	**1.441**	**0.785**
0.785	**1.441**	**0.915**	**0.915**	**0.915**	**1.441**	**0.785**
	0.785	**1.441**	**1.441**	**1.441**	**0.785**	
		0.785	**0.785**	**0.785**		

Figure 2.1.19: HI-STORM UMAX MPC-37 Sub-Design Heat Load for Short and Standard Fuel

(All storage cell heat loads are in kW)

Note that this figure shows the per cell heat load limit for storage. The total permissible aggregate heat load may be less than the sum of each individual cell heat load. See Table 2.1.8 for corresponding permissible aggregate heat load and the helium backfill option.

		0.829	**0.829**	**0.829**		
	0.829	**1.521**	**1.521**	**1.521**	**0.829**	
0.829	**1.521**	**0.966**	**0.966**	**0.966**	**1.521**	**0.829**
0.829	**1.521**	**0.966**	**0.966**	**0.966**	**1.521**	**0.829**
0.829	**1.521**	**0.966**	**0.966**	**0.966**	**1.521**	**0.829**
	0.829	**1.521**	**1.521**	**1.521**	**0.829**	
		0.829	**0.829**	**0.829**		

Figure 2.1.20: HI-STORM UMAX MPC-37 Sub-Design Heat Load for Long Fuel

(All storage cell heat loads are in kW)

Note that this figure shows the per cell heat load limit for storage. The total permissible aggregate heat load may be less than the sum of each individual cell heat load. See Table 2.1.8 for corresponding permissible aggregate heat load and the helium backfill option.

		0.97	0.97	0.97		
	0.97	0.97	0.97	0.97	0.97	
0.97	0.97	0.7	0.7	0.7	0.97	0.97
0.97	0.97	0.7	0.7	0.7	0.97	0.97
0.97	0.97	0.7	0.7	0.7	0.97	0.97
	0.97	0.97	0.97	0.97	0.97	
		0.97	0.97	0.97		

Figure 2.1.21: HI-STORM UMAX MPC-37 Permissible Threshold Heat Loads for VDS High Burnup Fuel

(All storage cell heat loads are in kW)

Note that this figure shows the per cell heat load limit for storage. The total permissible aggregate heat load may be less than the sum of each individual cell heat load. See Table 2.1.8 for corresponding permissible aggregate heat load and the helium backfill option.

				0.431	0.431	0.431				
		0.431	0.431	0.431	0.607	0.431	0.431	0.431		
	0.431	0.431	0.607	0.607	0.607	0.607	0.607	0.431	0.431	
	0.431	0.607	0.607	0.607	0.607	0.607	0.607	0.607	0.431	
0.431	0.431	0.607	0.607	0.431	0.431	0.431	0.607	0.607	0.431	0.431
0.431	0.607	0.607	0.607	0.431	0.431	0.431	0.607	0.607	0.607	0.431
0.431	0.431	0.607	0.607	0.431	0.431	0.431	0.607	0.607	0.431	0.431
	0.431	0.607	0.607	0.607	0.607	0.607	0.607	0.607	0.431	
	0.431	0.431	0.607	0.607	0.607	0.607	0.607	0.431	0.431	
		0.431	0.431	0.431	0.607	0.431	0.431	0.431		
				0.431	0.431	0.431				

Figure 2.1.22: HI-STORM UMAX MPC-89 Heat Load for Long-Term Storage
(All storage cell heat loads are in kW)

Note that this figure shows the per cell heat load limit for storage. The total permissible aggregate heat load may be less than the sum of each individual cell heat load. See Table 2.1.9 for corresponding permissible aggregate heat load and the helium backfill option.

				0.387	0.387	0.387				
		0.387	0.387	0.387	0.546	0.387	0.387	0.387		
	0.387	0.387	0.546	0.546	0.546	0.546	0.546	0.387	0.387	
	0.387	0.546	0.546	0.546	0.546	0.546	0.546	0.546	0.387	
0.387	0.387	0.546	0.546	0.387	0.387	0.387	0.546	0.546	0.387	0.387
0.387	0.546	0.546	0.546	0.387	0.387	0.387	0.546	0.546	0.546	0.387
0.387	0.387	0.546	0.546	0.387	0.387	0.387	0.546	0.546	0.387	0.387
	0.387	0.546	0.546	0.546	0.546	0.546	0.546	0.546	0.387	
	0.387	0.387	0.546	0.546	0.546	0.546	0.546	0.387	0.387	
		0.387	0.387	0.387	0.546	0.387	0.387	0.387		
				0.387	0.387	0.387				

Figure 2.1.23: HI-STORM UMAX MPC-89 Sub-Design Heat Load

(All storage cell heat loads are in kW)

Note that this figure shows the per cell heat load limit for storage. The total permissible aggregate heat load may be less than the sum of each individual cell heat load. See Table **2.1.9 for** corresponding permissible aggregate heat load and the helium backfill option.

				0.44	0.44	0.44				
		0.44	0.44	0.44	0.35	0.44	0.44	0.44		
	0.44	0.44	0.35	0.35	0.35	0.35	0.35	0.44	0.44	
	0.44	0.35	0.35	0.35	0.35	0.35	0.35	0.35	0.44	
0.44	0.44	0.35	0.35	0.35	0.35	0.35	0.35	0.35	0.44	0.44
0.44	0.35	0.35	0.35	0.35	0.35	0.35	0.35	0.35	0.35	0.44
0.44	0.44	0.35	0.35	0.35	0.35	0.35	0.35	0.35	0.44	0.44
	0.44	0.35	0.35	0.35	0.35	0.35	0.35	0.35	0.44	
	0.44	0.44	0.35	0.35	0.35	0.35	0.35	0.44	0.44	
		0.44	0.44	0.44	0.35	0.44	0.44	0.44		
				0.44	0.44	0.44				

Figure 2.1.24: HI-STORM UMAX MPC-89 Permissible Threshold Heat Load for VDS High Burnup Fuel
(All storage cell heat loads are in kW)

Note that this figure shows the per cell heat load limit for storage. The total permissible aggregate heat load may be less than the sum of each individual cell heat load. See Table 2.1.9 for corresponding permissible aggregate heat load and the helium backfill option.

2.2 HI-STORM UMAX VVM COMPONENTS AND ISFSI STRUCTURES

The VVM is engineered for below-grade storage of the MPCs for the duration of its design life, and is designed to withstand normal, off-normal, and extreme environmental phenomena as well as accident conditions of storage with appropriate margins of safety. A discussion of measures to ameliorate corrosion of VVM components is presented in Chapter 8.

The structural limit criteria imposed on the VVM Components are specified to comply with the provisions of 10CFR72, with an embedded large margin of safety. Table 2.4.1 provides the principal acceptance criteria applicable to the VVM Components. The specifications of the materials of construction for the load bearing and non-load bearing parts are provided in Table 2.6.2 along with their maximum permissible temperature for different conditions of storage.

The ISFSI Structures in a HI-STORM UMAX ISFSI are:

a. The Support Foundation Pad (SFP)

The structural requirements on the SFP are focused on providing a robust support to the CEC structure (for shear and compression) and to limit the long-term settlement of the SFP. The minimum structural design requirements on the SFP are provided in Table 2.3.2 and the licensing drawing in Section 1.5.

ACI-318(2005) is the prescribed Code for SFP design. As specified in ACI-318(2005), the applicable loads on the SFP are:

- Dead load (from the ISFSI pad, the loaded VVM, and the mass of soil above the SFP).
- Live load (from the loaded vertical cask transporter bearing on the ISFSI pad).
- Seismic load (the additional inertia load in excess of the dead weight, live load transmitted to the SFP from the loaded VVM and the transporter under the ISFSI's DBE event).
- Long-term settlement.

The load combinations for the HI-STORM UMAX structural analysis of the SFP pursuant to ACI-318(2005) are provided in Table 2.4.3.

Of the above loads, the effect of long-term settlement on the SFP is treated together with the Dead load. The standard approach to compute the long-term settlement is provided in [2.4.3].

In the structural qualification of the SFP, the loading from the seismic event is computed using the dynamic elastic modulus corresponding to the minimum shear wave velocity of the subgrade layers specified in Table 2.3.2.

b. The ISFSI Pad

The ISFSI Pad girdles the Container Shell and extends to the underside of the Container Flange to form a rain water-resistant interface, and has a slight slope to direct water away from the CEC. The principal functions of the ISFSI pad are to provide the riding surface for the loaded transporter and also to enable rainwater to be channeled away from the storage arrays and into the ISFSI storm drain system.

The minimum structural design requirements on the ISFSI pad are provided in Table 2.3.2 and the licensing drawing in Section 1.5. The applicable loads on the ISFSI pad are:

- Dead load (Self weight)
- Live load (Weight of a loaded cask transporter)
- Seismic Load (Inertia load from the concrete pad and the transporter under the ISFSI's DBE event)

The applicable load combinations for the structural analysis of the ISFSI pad pursuant to ACI-318(2005) are provided in Table 2.4.3.

Note that the ISFSI pad is not expected to settle relative to the SFP over their Design Life due to the use of CLSM [2.2.1] or lean concrete to refill the space between the SFP and the ISFSI pad (i.e., Space A in Figure 2.4.4).

The design of the ISFSI pad together with the lateral subgrade must also satisfy the allowable bearing capacity requirement of ACI 360R-06 [2.6.5] for slabs on grade. In particular, the total load imparted by the ISFSI pad on the lateral subgrade, including the live load and seismic load from the transporter, shall be less than 50 percent of the allowable bearing capacity thereof when the load is applied uniformly.

c. Enclosure Wall (optional)

The Enclosure Wall, if used to sequester the substrate under the ISFSI pad from the surrounding subgrade, serves to render each VVM group autonomous and, at sites with elevated water table and serves as a means to prevent water intrusion in the VVM subgrade space. The Enclosure Wall does not have a structural function. In other words, as shown in Chapter 3, the Enclosure Wall is not needed to maintain the physical stability of the subgrade under the ISFSI pad under a Design Basis Earthquake in the limiting condition where the adjacent space has been excavated down to the SFP for construction purposes. Likewise it is shown in Chapter 3 that a Design Basis Missile will not reach any CEC in the ISFSI if the missile were to strike laterally through the excavated space adjacent to the ISFSI.

d. Lateral Subgrade and Under-grade

The soil lateral to the CECs (termed Space A in this FSAR) is required to be removed and replaced with a Self-hardening Engineered Subgrade (SES) such as CLSM or lean concrete which imparts enhanced structural characteristics to the ISFSI pad support system improving its ability to support the Cask transporter during MPC transfer operations. The minimum average density and the minimum shear wave velocity in the lateral subgrade surrounding the VVMs have been specified in Table 2.3.2.

In addition, the design of the sub-grade under the ISFSI pad must be demonstrated to meet the following structural criteria under the Design Basis earthquake (Table 2.3.2) and the Design Basis Missile (Table 2.3.3) impact loading events:

a. The subgrade must continue to maintain its physical integrity under the DBE load combination in Table 2.4.3. Maintaining physical integrity means no structural collapse, instability or cracks that produce an unobstructed direct streaming path in the

subgrade for the radiation emanating from the stored fuel, and no constriction of the CEC that may interfere with retrievability of the MPC.

b. In the scenario where the adjacent subgrade has been excavated exposing the lateral surface of the subgrade and a Design Basis Missile (see Table 2.3.3) strikes the exposed surface in the most severe orientation, the sub grade must be capable of stopping the missile before it reaches the MPC.

The Under-grade's minimum properties have also been specified in Table 2.3.2. A structurally inadequate under-grade may be strengthened by suitably engineered pilings.

Version "MSE" which is solely focused on increasing the seismic capability of the "UMAX" system is described in Subsection 1.0.2. To meet increased seismic inertia loads, the strength of the material in the interstitial space between the VVMs (Space A in Figure 2.4.4) is increased by using normal density concrete. The minimum compressive strength of the Space A concrete is provided in Table 2.3.10

2.3 SERVICE CONDITIONS AND APPLICABLE LOADS

2.3.1 Service Conditions

The categories of loads applicable to the HI-STORM UMAX VVM are identified below.

Normal Condition: dead weight, handling of the Closure Lid, soil overburden pressure from subgrade, live load due to cask transporter movement, long-term settlement and snow loads.

Off-Normal Condition: elevated ambient temperature, wind, partial blockage of air inlets, and off-normal pressure.

Extreme Environmental Phenomena and Accident Condition: extreme ambient temperature, handling accidents, fire, tornado, flood, earthquake, explosion, lightning, complete (assumed) blockage of all inlet ducts, burial under debris, and 100% fuel rod rupture.

Short-term Operations: Short-term operation includes those normal operational evolutions necessary to support fuel loading or unloading activities. These include, but are not limited to MPC cavity drying, helium backfill, MPC transfer, and on-site handling of a loaded HI-TRAC VW transfer cask. The peak cladding temperature in the MPC must meet the ISG-11 Rev. 3 limit for short-term operations

Of the above loadings, lightning is considered to be innocuous to the HI-STORM UMAX ISFSI because of its underground configuration and is therefore not considered as a loading that merits safety evaluation.

As can be seen from the above, the loads that are most significant to the storage system's structures and components are either structural or thermal in origin. Accordingly, they are discussed in the two discrete sections in the following Sections 2.4 and 2.5. The design basis magnitudes of the above loads, as applicable to the HI-STORM VVM, are provided in Tables 2.3.1 and 2.3.2. The loads applicable to the MPC and HI-TRAC VM are defined in Tables 2.2.6, 2.2.7, 2.2.13 and 3.1.1 of the HI-STORM FW FSAR.

2.3.2 Loadings Applicable to Normal Conditions of Storage

2.3.2.1 Pressure

The MPC internal pressure is dependent on the initial volume of cover gas (helium), the volume of fill gas in the fuel rods, the fraction of fission gas released from the fuel matrix, the number of fuel rods assumed to have ruptured, and temperature.

The normal condition MPC internal design pressure bounds the cumulative effects of the maximum fill gas volume, normal environmental ambient temperatures, the maximum MPC heat load, and an assumed 1% of the fuel rods ruptured with 100% of the fill gas and 30% of the significant radioactive gases (e.g., H^3, Kr, and Xe) released in accordance with NUREG-1536.

For the storage of damaged fuel assemblies or fuel debris in a damaged fuel container (DFC), it shall be conservatively assumed that 100% of the fuel rods are ruptured with 100% of the rod fill gas and 30% of the significant radioactive gases (e.g., H^3, Kr, and Xe) liberated. For PWR

assemblies stored with non-fuel hardware, 100% of the gases in the non-fuel hardware (e.g., BPRAs) shall be assumed to be released. The accident condition design pressure shall envelop the case of 100% of the fuel rods ruptured.

The MPC internal and external pressures under the normal condition of storage must remain below the respective limits in docket number 72-1032. For convenience of reference, the maximum allowable internal and external pressures common to all MPCs in Table 1.2.1 are provided in Table 2.3.5.

The HI-TRAC transfer cask is not capable of retaining internal pressure due to its open design. Therefore, no analysis is required for the internal pressure loading in HI-TRAC VW transfer cask. However, the HI-TRAC transfer cask water jacket may experience an internal vapor pressure due to the heat-up of the water contained in the water jacket. Analysis is performed in Chapter 3 of the HI-STORM FW FSAR to demonstrate that the water jacket can withstand the design pressure in Table 2.3.5 without a structural failure and that the water jacket design pressure will not be exceeded. To provide an additional layer of safety, a pressure relief device is used to ensure that the water jacket design pressure will not be exceeded.

The HI-STORM UMAX VVM is not capable of retaining internal pressure due to its open design, and therefore no analysis is required or provided for the VVM internal pressure.

2.3.2.2 Environmental Temperatures and Pressures

To evaluate the long-term effects of ambient temperatures on the HI-STORM UMAX System, an upper bound value on the annual average ambient temperature for the continental United States is used. The annual average temperature is termed as normal ambient temperature for storage. The normal ambient temperature specified in Table 2.3.6 is bounding for all reactor sites in the contiguous United States. The normal ambient temperature set forth in Table 2.3.6 is intended to ensure that it is greater than the annual average of ambient temperature at any location in the continental United States. In the northern region of the U.S., the design basis normal ambient temperature will be exceeded only for brief periods, whereas in the southern U.S, it may be straddled daily in summer months. In as much as the sole effect of the normal temperature is on the computed fuel cladding temperature to establish long-term fuel integrity, it should not lie below the time averaged yearly mean for the ISFSI site. Previously licensed cask systems have employed lower normal temperatures (viz., 75^{o} F in Docket 72-1007) by utilizing national meteorological data.

Likewise, within the thermal analysis, a conservatively assumed soil temperature of the value specified in Table 2.3.6 is utilized to bound the annual average soil temperatures for the continental United States. The 1987 ASHRAE Handbook (HVAC Systems and Applications) reports average earth temperatures, from 0 to 10 feet below grade, throughout the continental United States. The highest reported annual average value for the continental United States is 77^{o}F for Key West, Florida. Therefore, this value is specified in Table 2.3.6 as the bounding soil temperature.

Confirmation of the site-specific annual average ambient temperature and soil temperature is to be performed by the licensee, in accordance with 10CFR72.212. Insolation based on 10CFR71.71 input averaged over 24 hours shall be used as the additional heat input under the normal and off-normal conditions of storage.

The ambient pressure shall be assumed to be 760mm of Hg coincident with the normal condition temperature, whose bounding value is provided in Table 2.3.6. For sites located substantially above sea level (elevation > 1500 feet), it will be necessary to perform a site specific evaluation of the peak cladding temperature using the site specific ambient temperature (maximum average annual temperature based on 40 year meteorological data for the site). ISG 11, Revision 3 [2.4.6] temperature limits will be applicable.

All of the above requirements are consistent with those in the HI-STORM 100 and HI-STORM FW FSARs.

2.3.2.3 Design Temperatures

The ASME Boiler and Pressure Vessel Code (ASME Code) requires that the value of the vessel design temperature be established with appropriate consideration for the effect of heat generation internal or external to the vessel. The decay heat load from the spent nuclear fuel is the internal heat generation source for the HI-STORM UMAX System. The ASME Code (Section III, Paragraph NCA-2142) requires the design temperature to be set at or above the maximum through thickness mean metal temperature of the pressure part under normal service (Level A) condition. Consistent with the terminology of NUREG-1536, this temperature is referred to as the Design Temperature for Normal Conditions. Conservative calculations of the steady-state temperature field in the HI-STORM UMAX System, under assumed environmental normal temperatures with the maximum decay heat load shall remain below the maximum permissible temperatures set down in Table 2.3.7. Unless otherwise stated, the maximum permissible temperatures in Table 2.3.7 are thru-wall thickness average values and are intended to insure that the storage system meets the safety criteria applicable to the specific operating condition.

Maintaining fuel rod cladding integrity is a principal design consideration. The fuel rod peak cladding temperature (PCT) limits for all operating conditions shall meet the limits set forth in ISG-11, Revision 3 [2.4.6].

2.3.2.4 Snow and Ice

The HI-STORM UMAX System must be capable of withstanding pressure loads due to snow and ice. Section 7.0 of ANSI/ASCE 7-05 [2.2.2] provides empirical formulas and tables to compute the effective design pressure on the VVM due to the accumulation of snow for the contiguous U.S. and Alaska. Typical calculated values for heated structures such as the HI-STORM UMAX System range from 50 to 70 pounds per square foot. For conservatism, the snow pressure load provided in Table 2.3.1 is set to bound the ANSI/ASCE 7-05 recommendation.

2.3.2.5 Dead Weight

The HI-STORM UMAX System must withstand the static loads due to the weights of each of its components, including the weight of the HI-TRAC VW with the loaded MPC stacked on top the VVM during the MPC transfer.

2.3.2.6 Handling Evolutions

The HI-STORM UMAX System must withstand loads experienced during routine handling. Normal handling includes:

i. Vertical lifting of HI-STORM UMAX Enclosure Lid.
ii. Vertical lifting and handling of the HI-TRAC VW transfer cask containing a loaded MPC.
iii. Lifting of a loaded MPC.

The dead load of the lifted component is increased by 15% in the stress qualification analyses (to meet ANSI N14.6 guidance) to account for dynamic effects from lifting operations as suggested in CMAA #70 [2.3.3].

2.3.2.7 Sustained Wind Conditions

Wind is an environment variable that can be characterized as a normal occurrence. However, it should not be confused with *normal condition of storage* as defined in NUREG-1536. Like the environmental temperature, the wind is a fickle quantity; it changes almost constantly. Normal condition of storage is based on the annual average of a site's ambient temperature, not daily, or weekly or some other short term average. A fixed velocity of wind (i.e., constant in magnitude, direction and sense of action) cannot be expected to persist long enough to allow the cask to reach steady state. To merit being considered a parameter for inclusion for normal condition of storage, its average for the whole year must be considered. Variation in the wind speed and direction, like the ambient temperature, is a fact of life. If the wind is reasonably constant at a site for some periods then it may be possible to identify a three-day average of the wind speed and also specify a fixed direction of action. But average of the wind velocity vector for a whole year for the normal storage case is evidently not meaningful or realistic. It is for the above reason that it is reasonable to consider three-day average wind velocity to be classified as a contributor to the off-normal condition which uses three day temperature average as the reference ambient temperature.

However, in the interest of conservatism, the fictitious annual wind average is treated as a design basis input to the normal condition of storage in this docket.

2.3.3 Loadings Applicable to Off -Normal Conditions of Storage

As the HI-STORM UMAX System is passive, loss of power and instrumentation failures are not defined as off-normal conditions. The off-normal service conditions are defined in this subsection.

A discussion of the effects of each off-normal condition and the corrective action for each off-normal condition is provided in Chapter 12.

2.3.3.1 Pressure

The HI-STORM UMAX System must withstand loads due to off-normal pressure. The off-normal condition for the MPC internal design pressure bounds the cumulative effects of the maximum fill gas volume, normal environmental ambient temperatures, the maximum MPC heat load, and an assumed 10% of the fuel rods ruptured with 100% of the fill gas and 30% of the significant radioactive gases (e.g., H^3, Kr, and Xe) released as suggested in NUREG-1536.

2.3.3.2 Environmental Temperatures

The HI-STORM UMAX System must withstand off-normal environmental temperatures. The off-normal environmental temperatures are specified in Table 2.3.6. For conservatism the lower bound temperature is assumed to occur with no solar loads and the upper bound temperature occurs with steady-state insolation. Each bounding temperature is assumed to persist for a sufficient duration to allow the system to reach steady-state temperatures.

Limits on the peaks in the time-varying ambient temperature at an ISFSI site are recognized in the FSAR in the specification of the off-normal temperatures. The lower bound off-normal temperature is defined as the minimum of the 72-hour average of the ambient temperature at an ISFSI site. Likewise, the upper bound off-normal temperature is defined by the maximum of 72-hour average of the ambient temperature. The lower and upper bound off-normal temperatures listed in Table 2.3.6 are intended to cover all ISFSI sites in the continental U.S. The 72-hour average of temperature used in the definition of the off-normal temperature recognizes the considerable thermal inertia of the HI-STORM UMAX storage system which essentially flattens the effect of daily temperature variations on the internals of the MPC.

2.3.3.3 Design Temperatures

In addition to the normal condition design temperatures, which apply to long-term storage conditions, an off-normal/accident condition temperature pursuant to the provisions of NUREG-1536 and Regulatory Guide 3.61 is also defined. This is the temperature which may exist during a transient event (examples of such an instance is the blockage of the VVM inlet vents or the fire accident). The off-normal/ accident condition temperatures of Table 2.3.7 are selected to bound the maximum (maximum in time and space) value of the thru-thickness average temperature of the structural or non-structural part, as applicable, during the transient event. These enveloping values, therefore, will bound the maximum temperature reached anywhere in the part, excluding skin effects, during or immediately after, a transient event.

The off-normal/accident condition temperatures for stainless steel and carbon steel components are chosen such that the material's ultimate tensile strength does not fall below 30% of its room temperature value, based on published data [2.4.8 and 2.4.13]. This ensures that the material will not be subject to significant creep rates during these short duration transient events.

Additionally, design temperature limits are also defined for short-term normal operating conditions which include but are not limited to MPC drying operations and onsite transport operations. The short-term temperature limits for all the components are specified in Table 2.3.7

2.3.3.4 Leakage of One Seal

The MPC enclosure vessel (in any MPC model listed in Table 1.2.1) does not contain gaskets or seals: All confinement boundary closure locations are welded. Because the material of construction (austenitic stainless steel) is known from extensive industrial experience to lend to high integrity, high ductility and high fracture strength welds, the MPC enclosure vessel welds provide a secure barrier against leakage.

The confinement boundary is defined by the MPC shell, MPC baseplate, MPC lid, port cover plates, closure ring, and associated welds. Most confinement boundary welds are inspected by radiography or ultrasonic examination. Field welds are examined by the liquid penetrant method on the root (if more than one weld pass is required) and final weld passes. In addition to multi-pass liquid penetrant examination, the MPC lid-to-shell weld is pressure tested. The vent and drain port cover plates are also subject to proven non-destructive evaluations for leak detection such as liquid penetrant examination. These inspection and testing techniques are performed to verify the integrity of the confinement boundary.

The HI-STORM UMAX VVM does not serve a confinement function: It does not feature any safety significant seals. Therefore, leakage of one seal is not evaluated for its consequence to the storage system.

2.3.3.5 Malfunction of FHD

The FHD system is a forced helium circulation device used to effectuate moisture removal from loaded MPCs. For circulating helium, the FHD system is equipped with active components requiring external power for normal operation.

Initiating events of FHD malfunction are: (i) a loss of external power to the FHD System and (ii) an active component trip. In both cases a stoppage of forced helium circulation occurs and heat dissipation in the MPC transitions to natural convection cooling.

Although the FHD System is monitored during its operation, stoppage of FHD operations does not require actions to restore forced cooling for adequate heat dissipation. This is because the condition of natural convection cooling evaluated in Section 4.6 of HI-STORM FW FSAR shows that the fuel temperatures remain below off-normal limits. An FHD malfunction is detected by operator response to control panel visual displays and alarms.

2.3.4 Extreme Environmental Phenomena and Accident Conditions

The loadings corresponding to the extreme environmental phenomena and accident events, collectively referred to as Faulted States, are discussed as a part of load combinations.

2.3.4.1 Partial Blockage of MPC Basket Flow Holes

The MPC is designed to prevent reduction of thermosiphon action due to partial blockage of the MPC basket flow holes by fuel cladding failure, fuel debris and crud. The HI-STORM UMAX System maintains the SNF in an inert environment with fuel rod cladding temperatures below accepted values (Table 2.3.7). Therefore, there is no credible mechanism for gross fuel cladding degradation of fuel classified as undamaged during storage in the HI-STORM UMAX. Fuel classified as damaged fuel or fuel debris are placed in damaged fuel containers. The damaged fuel container is equipped with mesh screens which ensure that the damaged fuel and fuel debris will not escape to block the MPC basket flow holes. The MPC is loaded once for long-term storage and, therefore, buildup of crud in the MPC due to numerous loadings is precluded. Using crud quantities for fuel assemblies reported in an Empire State Electric Energy Research Corporation Report [2.2.3] determines a layer of crud of conservative depth that is assumed to partially block the MPC basket flow holes. The crud depth is listed in Table 2.2.8 of the HI-STORM FW FSAR. The flow holes in the bottom of the fuel basket are designed (as can be seen on the licensing drawings) to ensure that this amount of crud does not block the internal helium circulation.

2.3.4.2 Confinement Boundary Leakage

None of the postulated environmental phenomenon or accident conditions identified will cause failure of the confinement boundary. Section 7.1 of the HI-STORM FW FSAR provides the rationale to treat leakage of the radiological contents from the MPC as a non-credible event.

2.3.4.3 Handling Accident

A handling accident in the Part 72 jurisdiction is precluded by the requirements and provisions specified in the HI-STORM FW FSAR. The loaded HI-TRAC VW and MPCs will be lifted in the Part 72 operations jurisdiction in accordance with written and Q.A. validated procedures and shall use special lifting devices which comply with ANSI N14.6-1993 [2.2.2]. Also, the lifting and handling equipment (typically the cask transporter) is required to have a built-in redundancy against uncontrolled lowering of the load. Further, the HI-TRAC VW is a vertically deployed system, and the handling evolutions in *short term operations*, as discussed in Chapter 9, do not involve downending of the loaded cask to the horizontal configuration (or upending from the horizontal state) at any time. In particular, the loaded MPC shall be lowered into the HI-STORM UMAX VVM or raised from it using the HI-TRAC VW transfer cask and a MPC lifting system designed in accordance with ANSI N14.6. Therefore, analysis of a handling accident event involving a HI-TRAC VW or MPC is not required.

2.3.4.4 Non-Mechanistic Tip-Over

Because the HI-STORM UMAX VVM is situated underground and cannot be moved, a tip-over event is not a credible accident for this design.

2.3.4.5 Tornado

The HI-STORM UMAX System must withstand pressures, wind loads, and missiles generated by a tornado. The prescribed design basis tornado and wind loads for the HI-STORM FW System are consistent with NRC Regulatory Guide 1.76 [2.4.9], ANSI 57.9 [2.4.10], and ASCE 7-05 [2.4.11]. Table 2.3.4 provides the wind speeds and pressure drops applicable to the HI-STORM UMAX System.

The continued integrity of the MPC confinement boundary, within the HI-TRAC VW transfer cask, under impact from tornado-generated missiles in conjunction with the wind loadings is demonstrated in Chapter 3 of the HI-STORM FW FSAR.

2.3.5 Short-Term Operations

Short-term operations and their safety considerations are discussed in Chapter 9.

Table 2.3.1

LOADS, CRITERIA, APPLICABLE REGULATIONS, REFERENCE CODES, AND STANDARDS FOR THE VVM

Type	Criteria or Value and Reference Location in the FSAR	Basis, Regulation and Reference Code/Standard
Life:		
Design Life	60 yrs	-
Service Life	100 yrs	-
Licensed Life	20 years	10CFR72.42(a) & 10CFR72.236(g)
Structural:		
Design & Fabrication Codes: Foundation Pad and ISFSI Pad	ACI-318(05)	10CFR 72.24
Unreinforced Concrete Stress Limits (Closure Lid)	Applicable Sections of ACI-318(05)	10CFR72.24(c)(4)
Structural Steel	Section 2.6	10CFR72.24(c)(4)
VVM Closure Lid Dead Weight:	Section 3.2	R.G. 3.61
Design Internal Pressure	Atmospheric	Ventilated Module
Thermal:		
Design Basis Heat Load	Governed by ISG-11, Rev. 3	The permissible heat load is limited by the requirement that the temperature of the fuel cladding and the internal pressure in the MPC do not exceed allowable limits under all thermally significant loadings listed in Section 2.5.
Maximum Design Temperatures:		
Closure Lid Concrete		
Through-Thickness Section Average (Normal)	Table 2.3.7	ACI 318(2005)
Through-Thickness Section Average (Off-Normal and Accident)	Table 2.3.7	ACI 318(2005)
Structural Steel	Table 2.3.7	ASME Code, Section II, Part D
Divider Shell Thermal Insulation	Heat transfer resistance per Table 4.2.7. Must be stable under long-term normal and short-term accident conditions.	N/A
Confinement:		
	N/A, Provided by MPC	10CFR72.128(a)(3) and 10CFR72.236(d) & (e)
Retrievability:		
Normal/Off-Normal	Retrieval of the contents.	10CFR72.122(f), (h), (1), & (l)
Accident	Retrieval of the canister	ISG-3
Criticality:		

Table 2.3.1

LOADS, CRITERIA, APPLICABLE REGULATIONS, REFERENCE CODES, AND STANDARDS FOR THE VVM

Type	**Criteria or Value and Reference Location in the FSAR**	**Basis, Regulation and Reference Code/Standard**
	Provided by MPC	10CFR72.124 and 10CFR72.128(a)(2)
Radiation Protection/Shielding:		
Normal/Off-Normal	Provide capability to meet controlled area boundary dose limits under 10CFR72 for all normal and off-normal conditions	10CFR72.104 and 10CFR72.212
	Ensure dose rates on and around the VVM during MPC transfer and lid installation operations are ALARA	10CFR20
Accident or Conditions of Extreme Environmental Phenomena	Meet controlled area boundary dose limits in regulations for all accidents	10CFR72.106
Design Bases:		
Spent Fuel Specification	N/A: Governed by the MPC's CoC with heat load adjusted per Section 2.1	10CFR72.236(a)
Normal Design Event Conditions:		
Ambient Outside Temperature:	-	-
Max. Yearly Average	Table 2.3.6	ANSI/ANS 57.9
Live Load†:		
Loaded HI-TRAC and Mating Device	Table 3.2.1	R.G. 3.61
Dry Loaded MPC	Table 3.2.1	R.G. 3.61
Cask Transporter	Table 3.2.1	-
Handling:		-
VVM Closure Lid Lift Points	Section 3.4	NUREG-0612 ANSI N14.6
Minimum Permissible Temperature During Closure Lid Handling Operations	10°F	ANSI/ANS 57.9
Snow and Ice Load	100 lb/ft^2	ASCE 7-88
Wet/Dry Loading	Dry	-
Storage Orientation	Vertical	-
Off-Normal Design Event Conditions:		
Ambient Temperature:		-
Minimum	Table 2.3.6	
Maximum	Table 2.3.6	
Partial Blockage of Air Inlets	50% blockage of air inlet	-

† Bounding weights.

Table 2.3.1

LOADS, CRITERIA, APPLICABLE REGULATIONS, REFERENCE CODES, AND STANDARDS FOR THE VVM

Type	Criteria or Value and Reference Location in the FSAR	Basis, Regulation and Reference Code/Standard
Design Basis Accident Events and Conditions:		
Drop Cases:		
End Drop	Not credible	In-ground VVM cannot be lifted
Tip-over	Not credible	In-ground VVM is constrained from tip over by ISFSI interfacing structures
Fire:	-	-
Amount of Fuel	50 Gallons	10CFR72.122(c)
Temperature	1475°F	10CFR72.122(c)
Fuel Rod Rupture	Chapter 4	-
Air Flow Blockage	100% blockage of air inlet flow area	10CFR72.128(a)(4)
Explosive Overpressure External Differential Pressure	10 psi steady state	10CFR72.128(a)(4)
Extreme Environmental Phenomenon Events and Conditions:		
Flood:		-
Height	125 ft	R.G. 1.59
Velocity	N/A	In-ground VVM is not subject to tip-over or sliding. Loads on the Closure Lid are bounded by missile impact loads.
Max. Earthquake	Table 2.3.2	10CFR72.102(f)
Tornado:	Table 2.3.4	-
Tornado-Borne Missiles:		
i. Automobile	Ensure shielding, fuel subcriticality and MPC retrievability and confinement	NUREG-1536
Weight	Table 2.3.3	NUREG-0800
Velocity	Table 2.3.3	NUREG-0800
ii. Rigid Solid Steel Cylinder (intermediate tornado missile)	Ensure shielding, fuel subcriticality and MPC retrievability and confinement	NUREG-1536
Weight	Table 2.3.3	NUREG-0800
Velocity	Table 2.3.3	NUREG-0800
iii. Steel Sphere	Section 2.4	NUREG-1536 In-ground VVM has no penetrations that provide line-of-sight to MPC
Weight	Table 2.3.3	NUREG-0800
Velocity	Table 2.3.3	NUREG-0800
Extreme Environmental Temp.	Table 2.3.6 (3-Day Average for the ISFSI Site)	-

Table 2.3.2

DESIGN DATA FOR HI-STORM UMAX ISFSI

	Type	Value (minimum or nominal, as applicable)	Comment
1.	ISFSI Pad and SFP concrete density concrete compressive strength rebar yield strength concrete cover on rebar	• 135 lb/ft^3 nominal dry density (ISFSI Pad) • 120 lb/ft^3 nominal dry density (SFP) • 4,500 psi minimum concrete compressive strength @ ≤ 28 days • 60,000 psi minimum rebar yield strength • minimum concrete cover on rebar per subsection 7.7.1 of ACI-318(05)	See Licensing Drawings in Section 1.5 for details on concrete pad thickness. Grade 60 Rebar. Rebar is #11@9" (each face, each direction)
2.	Depth averaged density of subgrade in Space A (see Figure 2.4.4)	120 lb/ft^3 minimum	Required for shielding and structural analysis
3.	Depth averaged density of subgrade in Space B (see Figure 2.4.4)	110 lb/ft^3 minimum	Required for shielding analysis.
4.	Depth averaged density of subgrade in Space C (see Figure 2.4.4)	120 lb/ft^3 nominal	Not required for shielding.
5.	Depth averaged density of subgrade in Space D (see Figure 2.4.4)	120 lb/ft^3 nominal	This space will typically contain native soil. Not required for shielding.
6.	Strain compatible effective shear wave velocity in Space A, V	1300 ft/sec minimum	This space will typically contain CLSM or lean concrete.
7.	Strain compatible effective shear wave velocity in Space B, V	450 ft/sec minimum	This space will typically contain native soil.

Table 2.3.2

DESIGN DATA FOR HI-STORM UMAX ISFSI

	Type	Value (minimum or nominal, as applicable)	Comment
8.	Strain compatible effective shear wave velocity in Space C, V	485 ft/sec minimum	This space may be remediated with vertical reinforcement such as pilings to enhance V.
9.	Strain compatible effective shear wave velocity in Space D, V (Figure 2.4.4)	485 ft/sec minimum	This space will typically contain native soil.
10.	Design Basis Earthquake	Top of the Grade (Ground surface) spectra per Figure 2.4.1 with horizontal ZPA, a_H and vertical ZPA, a_V scaled as follows: $a_H = 1.0g$ $a_V = 0.75g$ and foundation surface pad spectra per Figure 2.4.2 with horizontal ZPA, a_H and vertical ZPA, a_V of: $a_H = 0.93g$ $a_V = 0.71g$	Horizontal and vertical spectra shown in Figures 2.4.1 and 2.4.2 are based on 5% damping. Following the Newmark 100-40-40 response combination technique [2.6.7] endorsed by the Regulatory Guide 1.92 [2.4.7], the *resultant ZPA* for a 3-D earthquake site is defined as: $a_R = a_1 + 0.4a_2 + 0.4a_3$, where a_1, a_2 and a_3 are the site's ZPAs in three orthogonal directions and $a_1 \geq a_2 \geq a_3$. Hence, the DBE *resultant ZPAs* at ground surface and foundation surface elevations are 1.3 g's (=1.0×1.0g's + 0.4×0.75 g's) and 1.214 g's (=1.0×0.93g's + 0.4×0.71 g's), respectively.
11.	Permissible long-term settlement of the SFP	0.2 inch maximum	Used as the input value in the strength qualification of the SFP.
12	Density of plain concrete in the Closure Lid (nominal)	150 lb/cubic feet	Used in shielding calculations
13	Reference compressive strength of plain concrete in the Closure Lid	4,000 psi	Used in analysis of mechanical loadings on the Closure Lid
14	Minimum compressive strength of SES in Space A (see Figure 2.4.4)	1,000 psi	Used in tornado missile impact analysis and SSI analysis

Table 2.3.3

TORNADO-GENERATED MISSILES

Missile Description	**Mass (kg)**	**Velocity (mph)**
Automobile	1800	126
Rigid solid steel cylinder (8 in. diameter)	125	126
Solid sphere (1 in. diameter)	0.22	126

Table 2.3.4 CHARACTERISTICS OF REFERENCE TORNADO	
Condition	**Value**
Rotational wind speed (mph)	290
Translational speed (mph)	70
Maximum wind speed (mph)	360
Pressure drop (psi)	3.0

Table 2.3.5

DESIGN (MAXIMUM ALLOWABLE) PRESSURES

Pressure Location	**Condition**	**Pressure (psig)**
MPC Internal Pressure	Normal	100
	Off-Normal/Short-Term	110 (MPC-24, MPC-32 and MPC-68) 120 (MPC-37 and MPC-89)
	Accident	200
MPC External Pressure	Normal	(0) Ambient
	Off-Normal/Short-Term	(0) Ambient
	Accident	55
HI-TRAC Water Jacket Internal Pressure	Accident	65

Table 2.3.6

ENVIRONMENTAL TEMPERATURES

Condition	Temperature (ºF)	Comments
Normal Ambient Temperature (Bounding Annual Average from the contiguous United States)	80	
Normal Soil Temperature (Bounding Annual Average from the contiguous United States)	77	
Off-Normal Ambient Temperature (3-Day Average)	-40 (min) and 100 (max)	-40°F with no insolation 100°F with insolation
Extreme Accident Level Ambient (3-Day Average)	125	125°F with insolation
Short-Term Operations	0 (min.) 90 (max.)	The lower bound limit is specified in the technical specifications. The upper bound limit is a 3-day daily average with insolation and can be increased for a specific site if justified by the appropriate thermal analysis.

Table 2.3.7

DESIGN TEMPERATURES

Component	Normal & Mechanical Accident Condition Design Temperature Limits (°F) (Note 1)	Short-Term Events Temperature Limits (°F) (Note 6)	Off-Normal and Accident Condition Temperature Limits (°F) (Note 2)
MPC shell	650	800	1058
MPC basket	752	932	1058 (Note 5)
MPC basket shims	752	932	1058
MPC lid	752	800	1058
MPC closure ring	752	800	1058
MPC baseplate	752	800	1058
CEC shell	650	650	1058
CEC Flange	650	650	1058
Fuel Cladding	752 (Storage)	752 or 1058	1058 (Note 3)
Closure Lid concrete(section average)	350	350	600 (Note 4)
Closure Lid Top and Bottom Plate	650	650	1058
Remainder of VVM steel structure	650	650	1058
Divider Shell	650	650	1058
Insulation	650	650	1058
HI-TRAC VW inner shell	-	500	700
HI-TRAC VW bottom lid	-	350	700
HI-TRAC VW top flange	-	400	650
HI-TRAC VW bottom lid seals	-	350	N/A
HI-TRAC VW bottom lid bolts	-	350	800
HI-TRAC VW bottom flange	-	350	700
HI-TRAC VW radial neutron shield	-	311	N/A

Table 2.3.7

DESIGN TEMPERATURES

Component	Normal & Mechanical Accident Condition Design Temperature Limits (°F) (Note 1)	Short-Term Events Temperature Limits (°F) (Note 6)	Off-Normal and Accident Condition Temperature Limits (°F) (Note 2)
HI-TRAC VW radial lead gamma shield	-	350	600

Note1: Column 2 temperature limits apply to normal conditions of storage and to accident conditions that do not involve a thermally adverse condition such as seismic loading.

Note 2: For extreme environmental phenomena and accident conditions which involve heating of the steel structures and no mechanical loading (such as the blocked air duct accident, fire and burial-under-debris), the maximum permissible temperature is set equal to the permissible fuel cladding temperature(see Note 3 for MBF) at which the VVM structure will remain physically stable .

Note 3: Short term operations include but are not limited to MPC drying and onsite transport. The 1058°F temperature limit applies to MPCs containing all moderate burnup fuel. The limit for MPCs containing one or more high burnup fuel assemblies is 752°F.

Note 4: The VVM closure lid concrete, the primary function of which is shielding, will maintain its structural, thermal and shielding properties provided that the temperature limits prescribed.

The Portland Cement Association (PCA) Bulletin ST32 [2.3.1] provides a comprehensive assay of high temperature effects on concrete, stating, "Under certain conditions such as reinforced concrete chimneys, it (concrete) has been successfully used for temperatures of 600 deg. F (315.5°C). Where there is no load, such as chimney linings, it has been successfully used for temperature up to 1000 deg. F (537.8°C)".

The hydrogen loss from closure lid concrete due to high temperature is small because the closure lid concrete is completely enclosed by steel plate.

Note 5: The MPC basket contains stainless steel and Metamic (MPC-24, MPC-32 and MPC-68) or Metamic-HT (MPC-68M, MPC-37 and MPC-89). The neutron absorber material used in MPC baskets for criticality control (Metamic or Metamic-HT) are manufactured with B_4C and aluminum. B_4C is a refractory material that is unaffected by high temperature and aluminum is solid at temperatures in excess of 1200°F [2.3.2]. For conservatism, neutron absorbers temperatures under off-normal and accident conditions are limited to 1058°F.

Note 6: Short term operations include but are not limited to MPC drying and onsite transport. The 1058°F temperature limit applies to MPCs containing all moderate burnup fuel. The limit for MPCs containing one or more high burnup fuel assemblies is 752°F.

Table 2.3.8

MPC CONFINEMENT BOUNDARY STRESS INTENSITY LIMITS FOR DIFFERENT LOADING CONDITIONS (ELASTIC ANALYSIS PER NB-3220)†

Stress Category	Design	Level A	Level D††
Primary Membrane, P_m	S_m	S_m	A MIN ($2.4S_m$, $.7S_u$)
Local Membrane, P_L	$1.5S_m$	$1.5S_m$	150% of P_m Limit
Membrane plus Primary Bending	$1.5S_m$	$1.5S_m$	150% of P_m Limit
Primary Membrane plus Primary Bending	$1.5S_m$	N/A	150% of P_m Limit
Membrane plus Primary Bending plus Secondary	N/A	$3S_m$	N/A
Average Shear Stress††††	$0.6S_m$	$0.6S_m$	$0.42S_u$

† Stress combinations including F (peak stress) apply to fatigue evaluations only.

†† Governed by Appendix F, Paragraph F-1331 of the ASME Code, Section III.

†††† Governed by NB-3227.2 or F-1331.1(d).

Table 2.3.9

STRUCTURAL DESIGN CRITERIA FOR THE FUEL BASKET

PARAMETER	VALUE
Minimum service temperature	-40ºF
Maximum total (lateral) deflection in the active fuel region - dimensionless	0.005

Table 2.3.10

DESIGN DATA FOR HI-STORM UMAX VERSION MSE

(Properties listed in Table 2.3.2 remain unchanged except for those listed below, item numbers from Table 2.3.2 used where applicable)

No.	Type	Value (minimum or nominal, as applicable)	Comment
3.	Depth averaged density of subgrade in Space B (see Figure 2.4.4)	130 lb/ft^3	Required for shielding analysis and SSI analysis.
7.	Strain compatible effective shear wave velocity in Space B, V	344 ft/sec	A small value, computed using the site specific data at the San Onofre nuclear plant (documented in the Calculation Package [3.4.1]), used to magnify the inertial response of the system under the MSE.
10.	Design Basis Earthquake	Defined by Reg. Guide 1.60 spectra applied at the elevation of the Support Foundation Pad (SFP) (See Figures 2.4.5 and 2.4.6). The ZPAs of the two horizontal spectra, 1.5g's each, are vectorially combined to give the resultant ZPA of 2.12 g's. The vertical spectra is also Reg Guide 1.60 pegged at 1.0g.	Horizontal and vertical spectra shown in Figures 2.4.5 and 2.4.6 are based on 5% damping. Following the Newmark 100-40-40 response combination technique [2.6.7] endorsed by the Regulatory Guide 1.92 [2.4.7], the *resultant ZPA* for a 3-D earthquake site is defined as: $a_R = a_1 + 0.4a_2 + 0.4a_3$, where a_1, a_2 and a_3 are the site's ZPAs in three orthogonal directions and $a_1 \geq a_2 \geq a_3$. Hence, the DBE *resultant ZPA* at the foundation surface elevations is 2.5 g's (=1.0×1.5g's + 0.4×1.5 g's + 0.4x1.0 g's) This Newmark resultant ZPA is cited as the limiting value authorized for HI-STORM UMAX CoC for Version MSE.
14.	Minimum compressive strength of SES in Space A (see Figure 2.4.4)	3,000 psi	Used in tornado missile impact analysis and SSI analysis

2.4 STRUCTURALLY SIGNIFICANT LOAD COMBINATIONS AND ACCEPTANCE CRITERIA

Where appropriate, for each loading type, a bounding value is selected in this FSAR to impute an additional margin for the associated loading events. Such bounding loads are referred to as Design Basis Loads (DBL) in this FSAR. For example, the Design Basis External Pressure on the MPC, set down in Table 2.3.1, is a DBL, as it grossly exceeds any credible external pressure that may be postulated for an ISFSI site.

The Design Basis structural loads and their combinations applicable to normal, off-normal and accident conditions for the HI-STORM UMAX system are considered in this section and summarized in Tables 2.4.1 and 2.4.3. The qualifying analyses are presented in Chapter 3.

Each loading case in Table 2.4.1 is distinct in respect of the sub-component of the VVM that it affects most significantly. The acceptance criteria for the storage system, pursuant to NUREG-1536, consist of demonstrating that:

a. The radiation shielding in the storage system does not degrade under normal and off-normal conditions of storage.

b. The system does not deform under credible loading conditions in a manner that would jeopardize the subcritical state of the storage system or ready retrievability of the MPC.

c. The MPC maintains confinement of radiological matter. For accident condition loadings, any permissible degradation in shielding must be shown to result in dose rates sufficiently low to permit recovery of the MPC from the damaged VVM, including unloading if necessary, and loss of function must be readily discernible ,i.e., apparent or detectable.

The above overarching acceptance criteria are further particularized in a more conservative form for each applicable loading case explicitly listed in Table 2.4.1 in the following sections.

2.4.1 Load Case 01: Dead Load plus Design Basis Explosion Pressure

The HI-STORM UMAX system must withstand the pressure pulse due to a design basis explosion event. The effect of overpressure due to an explosion near the VVM acting concurrently with the dead load of the system is defined as Load Case 01 and analyzed in Chapter 3. The overpressure design value applied to the Closure Lid outer surface (see Table 2.3.1) is intended to bound all credible explosion events because no combustible material is permitted to be stored near the VVM, and all materials of construction are engineered to be compatible with the operating environment. However, site-specific explosion scenarios that are not evidently bounded by the design basis explosion load considered herein shall be evaluated under the provisions of 10CFR72.212.

The explosion load is stated in Table 2.3.1 in terms of an equivalent static pressure. The affected sub-components are:

a. The Container Shell, subjected to a compressive state of axial stress under the combined effect of dead weight of the Closure Lid and surface pressure on the Closure Lid under the explosion event.

a. The Closure Lid, subject to self-weight and the Closure Lid surface pressure under the explosion event.

Other VVM Components are not in the direct path of this loading. Level D stress limits are applicable to this load case based on applicable metal service temperatures. The explosion pressure also envelops other normal condition mechanical loads such as snow and flood. Load Case 01, therefore, is a bounding load combination that conservatively subsumes other loading types. Acceptance criteria for this load case are provided in Table 2.4.1. Level A stress limits are applicable based on reference metal temperatures that bound all mechanical loading scenarios (Table 2.4.1) when this case is used as an enveloping evaluation for any normal condition.

2.4.2 Load Case 02: Design Basis Missile Loadings

The HI-STORM UMAX System is protected from the effects of a tornado and accompanying missiles by virtue of its underground configuration. The only VVM component that warrants evaluation for the effects of a tornado-induced missile strike is the Closure Lid, which is made of a steel weldment with encased concrete. The prescribed design basis tornado and wind loads for the HI-STORM UMAX System are consistent with NRC Regulatory Guide 1.76 [2.4.9], ANSI 57.9 [2.4.10] and ASCE 7.05 [2.4.11]. Design Basis Missiles are summarized in Table 2.3.3. The HI-STORM UMAX System is inherently stable under tornado missile impact. The impact of a large missile (1800kg Automobile) is evaluated to determine whether the Closure Lid continues to maintain its required shielding function. Penetration and perforation issues associated with the Closure Lid due to intermediate missiles that constitute the Extreme Environmental Phenomena loads for the HI-STORM UMAX system are also addressed. The Closure Lid is analyzed for penetration of a solid steel cylinder traveling at a high speed consistent with the characteristics of the intermediate missile listed in Table 2.3.3. As there is no direct line of sight to the MPC, small missiles are not considered. Also, since a tornado is a short duration event, the effect of tornado winds on the thermal performance of the VVM would be negligible due to the system's thermal inertia. Therefore, the effect of tornado wind on the thermal performance of the HI-STORM UMAX system is not analyzed.

When subject to a tornado missile strike, the Closure Lid must not be dislodged creating a direct line of sight from the top of the MPC to the outside (see Table 2.4.1). For the intermediate missile, the Closure Lid must resist full penetration. Finally, any CEC deformation from the compressive axial impulse due to the missile strike must not prevent MPC retrievability.

2.4.3 Load Case 03: Design Basis Seismic Event and Long-Term Settlement

The DBE and Long-Term Settlement are treated together under Load Case 03 because they are both germane to the qualification of the two pads (SFP and the ISFSI Pad).

a. Design Basis Seismic Event

As required by 10CFR72.102(f), the Design Basis Earthquake for the ISFSI must be specified. For the HI-STORM UMAX system, a generic Design Basis Earthquake is specified with horizontal and vertical ZPAs intended to envelope the site-specific DBEs at all U.S. plant sites (See Table 2.3.2). For purposes of the generic seismic analysis in this FSAR, the Design Basis Earthquake for the HI-STORM UMAX system is defined by two sets of response spectra specified at the ISFSI pad top surface elevation and at the SFP bottom surface elevation, as shown in Figures 2.4.1 and 2.4.2, respectively. These two spectra sets together exhibit the severity of the earthquake experienced by the ISFSI Structures and VVM Components and are henceforth referred to as the governing spectra. The two sets of response spectra are obtained from the two-step SHAKE/LS-DYNA soil seismic response analyses performed using a lower-bound soil shear wave velocity profile (see Figure 2.4.3). This lower bound profile was established in [2.4.1] based on the geotechnical data of typical U.S. nuclear power plant sites. To develop the governing spectra, the input seismic acceleration time history for the SHAKE analysis is derived from the Regulatory Guide 1.60 seismic response spectrum and designated as the rock outcrop motion. The synthetic time history complies with the response spectrum and power density enveloping criteria in SRP 3.7.1 in NUREG-0800, Rev 2. The input acceleration time history is scaled to yield ground surface ZPAs (at the top of grade elevation) as specified in Table 2.3.2. The average strain-compatible shear wave velocities of the soil column obtained from the SHAKE analysis are used to specify the minimum shear wave velocity values in Table 2.3.2.

The soil model for the subsequent LS-DYNA seismic response analysis uses the average strain-compatible wave velocities obtained from the SHAKE analysis (i.e., minimum shear wave velocity values of the native soils in Table 2.3.2) to define the structural characteristics of the soil layers above and below the SFP elevation. The acceleration time history at the soil column bottom surface, also obtained from the above-mentioned SHAKE analysis, is used as the input seismic motion for the LS-DYNA seismic response analysis performed in Chapter 3. The response spectrum plots shown in Figure 2.4.1 and 2.4.2 are the results of the LS-DYNA soil seismic response analysis (in the absence of the ISFSI). The same soil model and input seismic motion used in the LS-DYNA seismic response analysis is used for the LS-DYNA Soil-Structure Interaction (SSI) analysis (with the ISFSI included in the model) in Chapter 3.

The combination of weak soil properties and strong earthquake, as specified in Table 2.3.2 for the structural evaluation of the underground ISFSI, has been selected to ensure that the Design Basis Earthquake response spectra at the ISFSI location will uniformly envelope those at most U.S. nuclear plants and that the Design Basis structural evaluation for the HI-STORM UMAX system is performed conservatively based on the lower bound support from the sub-grade and the under-grade. Thus, the HI-STORM UMAX system can be deployed in most U.S. nuclear power plant sites without the need for a site-specific analysis to satisfy the requirements of 72.212. Specifically, a candidate HI-STORM UMAX ISFSI site will be exempt from a detailed SSI

analysis if the soil seismic response analysis for the site (using SHAKE or similar program) can demonstrate that the following two criteria are met:

- The site's response spectra at both ISFSI pad and SFP elevations are enveloped by the Design Basis Earthquake response spectra shown in Figures 2.4.1 and 2.4.2 at their applicable elevations, respectively;
- The soil properties of the candidate site are greater than the minimum values specified in Table 2.3.2.

In order to satisfy the first criterion, the site must perform a soil seismic response analysis using SHAKE or a similar program. The site's response spectra at both the ISFSI pad and SFP elevations must be bounded by the Design Basis Earthquake response spectra in Figures 2.4.1 and 2.4.2.

For the case where only one of the above two criteria is not satisfied, a site-specific evaluation under 10CFR72.212 is permitted. Typical scenarios that warrant a site specific evaluation are discussed below:

<u>Scenario A</u>: The site's response spectra are not completely enveloped by the respective Design Basis Earthquake response spectra in Figures 2.4.1 and 2.4.2. However, the site's overall earthquake strength, represented by the resultant ZPA (see Table 2.3.2 for definition) is bounded by that of the Design Basis Earthquake at both ISFSI pad and SFP elevations.

While the ZPA represents the strength of the earthquake (in terms of the maximum value of the seismic acceleration time history), the shape of the seismic response spectrum is affected by many factors such as the overall stiffness of the site and the stiffness profile of soil layers. Therefore, for the same input seismic time history at the base of the soil column, a stiffer site could have a peak response that is not enveloped by the Design Basis Earthquake response spectrum (as demonstrated in the SHAKE parametric study results presented in Table 2.4.4, where the only difference between the two analyzed cases is the stiffness (i.e., shear wave velocity) of the soil column). Although it is expected that the HI-STORM UMAX system would exhibit a greater safety margin against the earthquake loading at the stiffer subgrade/under-grade site, a site-specific evaluation under 10CFR72.212 is the appropriate vehicle to confirm the structural integrity in this situation.

<u>Scenario B</u>: The strain compatible wave velocity of the soil in Space B and/or Space D of the ISFSI site (see Figure 2.4.4) is less than the required minimum value specified in Table 2.3.2.

Typically, Spaces B and D (in Figure 2.4.4) contain native soils whose properties are not affected by the ISFSI construction. More importantly, the loaded VVMs are not directly supported by the soil in these two spaces. Therefore, it is reasonable to assume that a small reduction of soil stiffness in these two spaces would not significantly modify the structural response of the VVM system. Structural compliance through a site specific analysis is assured if the ZPA of the DBE is well below the Design Basis value set down in this FSAR (Figure 2.4.1 and 2.4.2).

The site-specific safety analysis, if required, shall follow the methodology set down in Chapter 3. In addition, since the soil and rock configuration varies from site to site, the total depth of the soil model for site-specific analysis shall be determined following the guideline in Paragraph 3.3.3.2 of ASCE 4-98 [2.4.3]. Uncertainties in SSI analysis for a candidate HI-STORM UMAX

ISFSI site shall be accounted for by varying the best estimate low strain shear modulus of the substrates between the best estimate values times (1+c) and the best estimate value divided by (1+c). If sufficient, adequate soil investigation data is available, the mean and standard deviation of the low strain shear modulus shall be established for every soil layer. The value of c may be established so that it will cover the mean plus or minus one standard deviation for every layer; however, the minimum value for c shall be no less than 0.5. If sufficient data is not available to determine a statistically meaningful mean and standard deviation, then the value for c shall be no less than 1.0.

The qualification of the ISFSI under the system's DBE event involves the following safety determinations:

a. Compliance of the VVM Components to the applicable stress/deformation limits specified in Table 2.4.2.

b. Strength compliance of the ISFSI reinforced concrete structures under ACI-318(2005) load combinations listed in Table 2.4.3.

The Design Basis Seismic Event (also referred to as the Design Basis Earthquake (DBE)) is classified as an extreme environmental phenomenon. As such the Level D service condition limits are applicable to the VVM components, such as the MPC Enclosure Vessel, MPC Guides and the MPC shell (Table 2.4.2).

Most Severe Earthquake Applicable to Version "MSE" of HI-STORM UMAX

As explained in Subsection 1.0.2, Version MSE, is strictly focused on further strengthening the VVM structure, where necessary, such that the system can meet the "most severe earthquake" defined by Regulatory Guide 1.60 spectra as set forth in Table 2.3.10. The required changes, as described in Subsection 1.0.2 and shown in the licensing drawings, are quite minimal. The MSE is designated as the "Design Basis Earthquake" for HI-STORM UMAX Version MSE.

Because the seismic qualification of the reference ISFSI under the MSE entails non-linear time history analysis, five sets of statistically independent time histories corresponding to the horizontal and vertical target spectra (Figures 2.4.5 and 2.4.6) [2.4.14] were generated to comply with NUREG 0800, SRP 3.7.1. Table 2.4.5 provides a comparison of the ZPAs of the generated time histories with those of the corresponding target spectra. As can be seen, the generated spectra bound both the ZPAs and peak values of target spectra.

Table 2.4.5 also provides the duration of the earthquake for each time history set. Figures 2.4.7 to 2.4.11 provide the accelerograms corresponding to the five earthquakes generated to perform the seismic analysis. The qualifying analyses under the above earthquakes are summarized in subparagraph 3.4.4.1.2.

Acceptance Criteria Under DBE

The CEC shell is subject to performance-based limits, which require that the deformation of the CEC does not prevent MPC retrievability, does not cause loss of MPC confinement, and that the system remains subcritical. This is accomplished by demonstrating that after the seismic event, permanent ovalization of the Container Shell does not result in a geometry that precludes retrievability of the MPC and that the impact loadings on the MPC due to its rattling inside the CEC do not cause a breach of the MPC confinement boundary.

Finally, because the MPC Enclosure Vessel is designed to meet ASME Section III, Subsection NB (Class 1) stress intensity limits, and the earthquake is categorized as a Level D event, the primary stress intensities in the MPC Enclosure Vessel must meet Level D limits. The primary stress intensity in the MPC shell is the maximum longitudinal flexural stress intensity, which is compared against the primary membrane stress intensity limit for the material (Alloy X) at the applicable service temperature.

The limits on the primary stresses in the MPC confinement boundary for the DBE condition are also applicable to other Level D (faulted) events. Dynamic analysis using a 3-D detailed model of the MPC confinement boundary is the vehicle for performing the structural qualification. In addition to the primary stress limits, the local plastic strain in the Confinement Boundary due to the impact between the MPC and the MPC guides under the Design Basis Earthquake requires evaluation.

b. Long-Term Settlement

At an ISFSI site, depending on the density of the subgrade, there may be a small mismatch between the weight of the excavated native soil above the SFP and the total weight of loaded VVMs and the refilled subgrade, leading to a minor amount of long-term settlement of the SFP. The limiting allowable value of the SFP long-term settlement has been specified in Table 2.3.2 for a conservative stress analysis of the pad under the load combinations of Table 2.4.3. The effect of long-term settlement on the SFP shall be considered as a concurrent load with Dead load. On the other hand, the ISFSI pad is founded on the refilled Self-hardening Engineered Subgrade (i.e., CLSM or lean concrete), which is known to be immune from long term settlement due to its own weight and the dead weight of the ISFSI pad.

2.4.4 Load Case 04: Design Basis Handling and Impact Events

Because the VVM is situated underground and cannot be moved, drop and tip-over events are not credible accidents for this design. The Closure Lid, as can be inferred from the Licensing Drawings, cannot strike the MPC lid if it were to undergo a free fall due to geometry constraints. Further, because the load handling device and lifting equipment are required to meet the defense-in-depth criteria set down in this FSAR, the drop of the Closure Lid or HI-TRAC transfer cask during handling operation is termed non-credible (as is the case for the aboveground HI-STORM system MPC transfer operations at the ISFSI).

The design of the lifting equipment must meet single failure proof criteria to preclude a safety related load handling event during emplacement or removal of the Closure Lid while the CEC contains a loaded MPC. The Closure Lid lifting attachments shall meet the strength limits of ANSI N14.6 for heavy load handling. The metal load bearing parts shall satisfy the requirements of Reg. Guide 3.61 for primary stresses near the lifting locations and shall satisfy ASME NF [2.6.1] Level A limits away from the lifting locations.

Yield and ultimate strength values used in the stress compliance demonstration per ANSI N14.6 shall utilize confirmed material test data through either independent coupon testing or material suppliers' CMTR or CoC, as appropriate.

Lift locations for the CEC are expected to be used for lifting only during construction, and possibly during decommissioning of the VVM with no loaded MPC present; therefore, these lifting locations are not subject to the defense-in-depth measures of NUREG-0612. They are therefore considered as a part of the site construction safety plan, and the site decommissioning plan, as applicable.

2.4.5 Load Case 05: Design Basis Fire Event

The potential of a fire accident near an ISFSI pad is considered to be rendered extremely remote by ensuring that there are no significant combustible materials in the area. The only credible concern is related to a transport vehicle fuel tank fire engulfing the loaded HI-STORM module during the MPC transfer operations or loaded HI-TRAC transfer cask while it is being moved to the ISFSI.

The VVM must withstand the effects of a fire that consumes the maximum volume of fuel permitted to be in the fuel tank of the cask transporter. The duration of the fire for the VVM is conservatively assumed to be the same as that used for the modules in docket numbers 72-1014 and 72-1032. As is the case for aboveground VVMs, the fuel is assumed to spill, surround one storage system and burn until it is depleted. Because the VVM is configured to have a surrounding built-in step or spill barrier, the spilled fuel will collect and burn over the ISFSI pad, also referred to as Top-of-Grade. Therefore, the location of fuel combustion will be physically removed from the CEC. Also, the natural grade in the ISFSI pad surface, engineered to direct the rainwater away from the VVMs, will do the same to the spilled fuel, further ameliorating the thermal consequence of the fire to the stored MPCs.

The sole effect of fire on the VVM structure is to raise the metal temperature of the structural members surrounding the shielding concrete in the Closure Lid. The analysis for the fire event accordingly seeks to establish that the load bearing structure will not be weakened by the rise in its metal temperature (and a consequent reduction in the yield and ultimate strength) and the Closure Lid suffers structural failure or instability.

Therefore, it is required to demonstrate that the structural collapse of the Closure Lid cannot occur due to the reduction of its structural material's (low carbon steel) strength at the elevated temperatures from the fire.

Finally, it is necessary to demonstrate that the internal pressure in the stored MPC will not exceed the accident condition design pressure during or after the fire event.

2.4.6 Load Case 06: Live Load on VVM During MPC Transfer

The VVM must withstand the weight of the loaded HI-TRAC transfer cask and the mating device during MPC transfer operations. Bounding weights for these components are used in the qualifying analysis.

The acceptance criterion for this load case, like all other load cases, is provided in Table 2.4.1.

2.4.7 Load Case 7: Design Basis Flood

The HI-STORM UMAX System is engineered to be flood resistant. Furthermore, the potential water ingress passages are elevated in the HI-STORM UMAX (in contrast to the pad level inlet ducts in typical ventilated VVMs) to prevent intrusion of floodwater in the MPC storage cavities. However, all HI-STORM MPCs are designed to withstand 125 feet of water submergence (Table 2.4.1). The VVM will clearly withstand this static head of water above the surface of the ISFSI because all structural members are either not subject to any pressure differential from the flood or are backed by the subgrade, which resists the flood water directly. Full or partial submergence of the MPC is not a concern from a thermal perspective, as discussed in Chapter 1, because heat removal is enhanced by the floodwater. Submergence of the CEC up to the level of the Container Flange will not result in a significant hydrostatic pressure; therefore, collapse of the CEC or potential for significant stresses is not a concern. Furthermore, the uplift force on the CEC due to buoyancy is less than the downward force due to the system's dead weight; therefore, uplift of the CEC is not a concern.

The analysis of the CEC for submergence under 125 feet of water is presented in Chapter 3.The qualification of the MPC under the Design Basis Flood is documented in the FSAR that supports the CoC of the MPC.

Table 2.4.1

LOAD CASES AND ACCEPTANCE CRITERIA

Load Case I.D.	Bounding Loading	Affected Sub-Component	Applicable Data: Magnitude of Loading	Applicable Data: Reference Coincident Metal Temperature (Deg. F)	Acceptance Criterion
01	Normal operation condition; dead load plus design basis explosion pressure	• Container Shell structure • Closure Lid	Section 3.2 (dead weight);Table 2.3.1 (pressure)	150 350	Primary stresses do not exceed applicable Level A stress limits of ASME Subsection NF for dead weight only (or Level D limits with explosion)
02	Design basis missile	Closure Lid	Table 2.3.3	350	Closure Lid does not collapse, is not dislodged from the cavity, and is not perforated by the missile.
03	Design basis earthquake	Container Shell	Figure 2.4.1 and 2.4.2	150	After the DBE event, MPC retrievability, subcriticality and confinement must not be compromised. Additional criteria for the CEC and its contents are defined in Table 2.4.2
04	Closure lid handling	Lid Lift Lugs; all metal structure in Lid	1.15 x Closure Lid Weight (From Table 3.2.1)	200	ANSI N14.6 limits based on yield or ultimate strength including magnified inertia loads. Meet Reg. Guide 3.61 and Level A limits as applicable.
05	Design basis fire	Closure Lid	Section 2.4.5	800	The Closure Lid structure does not collapse under its dead weight due to elevated metal temperatures.

Table 2.4.1

LOAD CASES AND ACCEPTANCE CRITERIA

Load Case I.D.	**Bounding Loading**	**Affected Sub-Component**	**Applicable Data**		**Acceptance Criterion**
			Magnitude of Loading	**Reference Coincident Metal Temperature (Deg. F)**	
06	VVM loaded by the overhead transfer cask and Mating Device during the transfer operation	Container Shell	-	150	Service A stress limit for NF Class 3 plate and shell structure and buckling stress limits for the VVM shell must be met.
07	Design Basis Flood	Container Shell	125 feet of water head	150	The hoop stress in the Container Shell shall be below the minimum material yield strength without taking credit for the action of the surrounding subgrade.

Note 1. Structural loads and acceptance criteria for each load case are further explained in Section 2.4.

Note 2: Materials of construction are identified in Table 2.6.2.

Note 3: Design attributes of the VVM are explained in Chapter 1 and details are presented in the drawings in Section 1.5.

Note 4: The limiting value of coincident metal temperature is used to establish material properties and allowable stress (or stress intensity) when applicable.

Note 5: Load cases applicable to MPC and HI-TRAC VW are presented in Tables 2.2.6, 2.2.7, 2.2.13 and 3.1.1 of the HI-STORM FW FSAR.

Table 2.4.2

ACCEPTANCE CRITERIA FOR THE HI-STORM UMAX VVM AND INTERNALS UNDER EXTREME ENVIRONMENTAL CONDITIONS

Component	Calculated Value	Allowable Limit
CEC Container Shell	Radial gap between Divider Shell and the MPC after the seismic event	Nominal Gap between the MPC and the Divider Shell must remain open at end of event.
MPC Guides	Maximum compressive load	Minimum of limiting buckling load or ultimate load
MPC Shell	Longitudinal flexural stress intensity in shell wall from bending of the MPC shell as a beam. The local true strain in the MPC shell in the region of MPC guide/MPC Top Lid impact.	ASME Level D primary membrane stress intensity limit The local strain from impact must be less than 10%, which has been established as a conservative limit in [2.4.12].

Table 2.4.3

LOAD COMBINATIONS FOR THE ISFSI PAD AND SUPPORT FOUNDATION PAD PER ACI-318 (2005)

Load Combination Case	Load Combination
LC-1	1.4D
LC-2	1.2D + 1.6L
LC-3	1.2D + E + L

where:

D: Dead Load including long-term settlement effects.

L: Live Load

E: DBE for the Site

Table 2.4.4

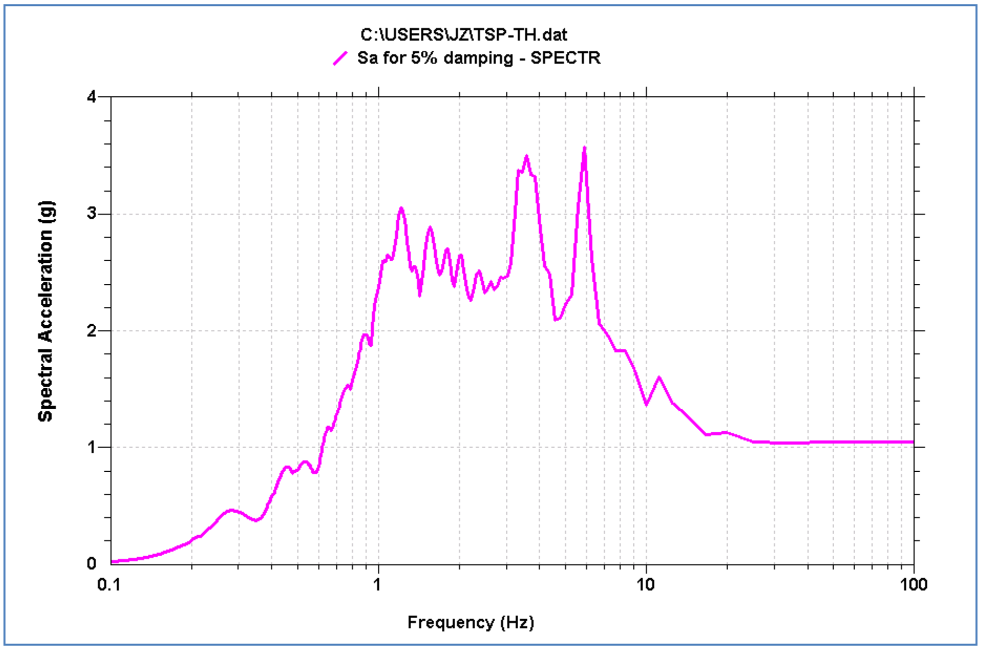
SHAKE PARAMETRIC STUDY OF THE EFFECT OF SUBGRADE PROPERTIES ON SOIL RESPONSES AT HI-STORM UMAX ISFSI TOP & BOTTOM ELEVATIONS

Elevation & Direction	**Acceleration Response**	**Value (g's)**	
		Lower Bound Shear Wave Velocity Profile (see Figure 2.4.3)	**Upper Bound Shear Wave Velocity Profile (see Figure 2.4.3)**
ISFSI pad Top Surface Horizontal Direction	ZPA	1.008	0.897
	Peak	3.851	4.040
SFP Bottom Surface Horizontal Direction	ZPA	0.930	0.795
	Peak	3.519	3.762
ISFSI pad Top Surface Vertical Direction	ZPA	0.751	0.539
	Peak	3.912	2.377
SFP Bottom Surface Vertical Direction	ZPA	0.706	0.516
	Peak	3.573	2.286

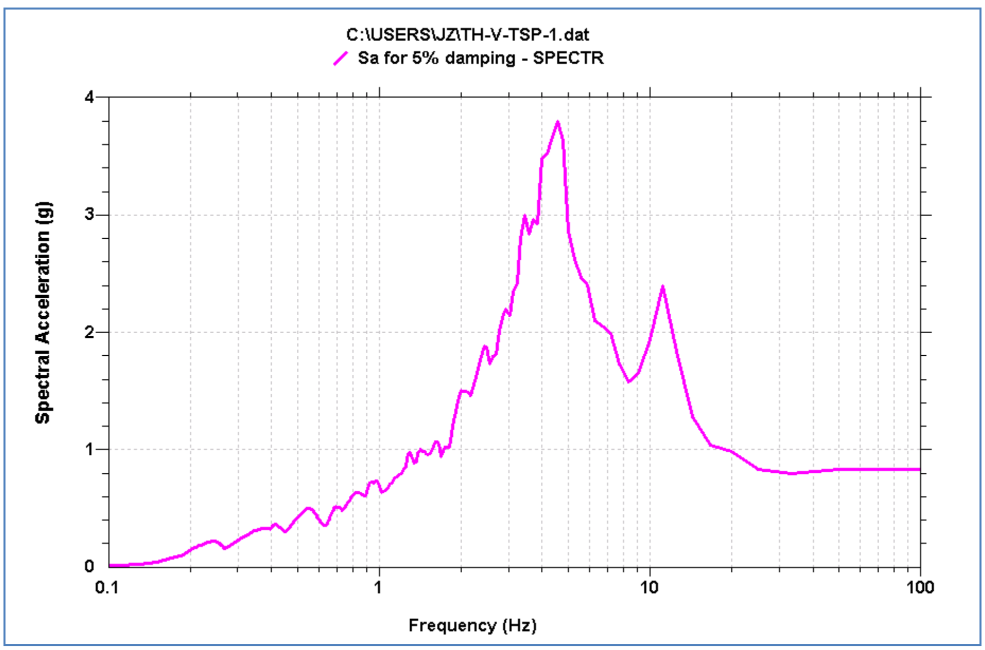
Table 2.4.5

COMPARING SPECTRA CORRESPONDING TO THE GENERATED TIME HISTORY SETS FOR " UMAX" VERSION MSE WITH THE TARGET SPECTRA (FIGURES 2.4.5 AND 2.4.6)

Item	Horizontal ZPA	Horizontal Peak	Vertical ZPA	Vertical Peak	Comment	Duration of the generated time history (sec)
Target Spectrum	**2.12**	**6.64**	**1.0**	**2.98**	**See Figures 2.4.5 and 2.4.6**	**N/A**
Time History Set (THS) #1	2.199	7.38	1.087	3.16	See accelerograms in Figure 2.4.7	40
THS #2	2.290	7.18	1.019	3.14	See accelerograms in Figure 2.4.8	39
THS#3	2.162	7.10	1.072	3.21	See accelerograms in Figure 2.4.9	44
THS#4	2.193	7.23	1.151	3.08	See accelrograms in Figure 2.4.10	50
THS #5	2.132	6.93	1.111	3.25	See accelerograms in Figure 2.4.11	90



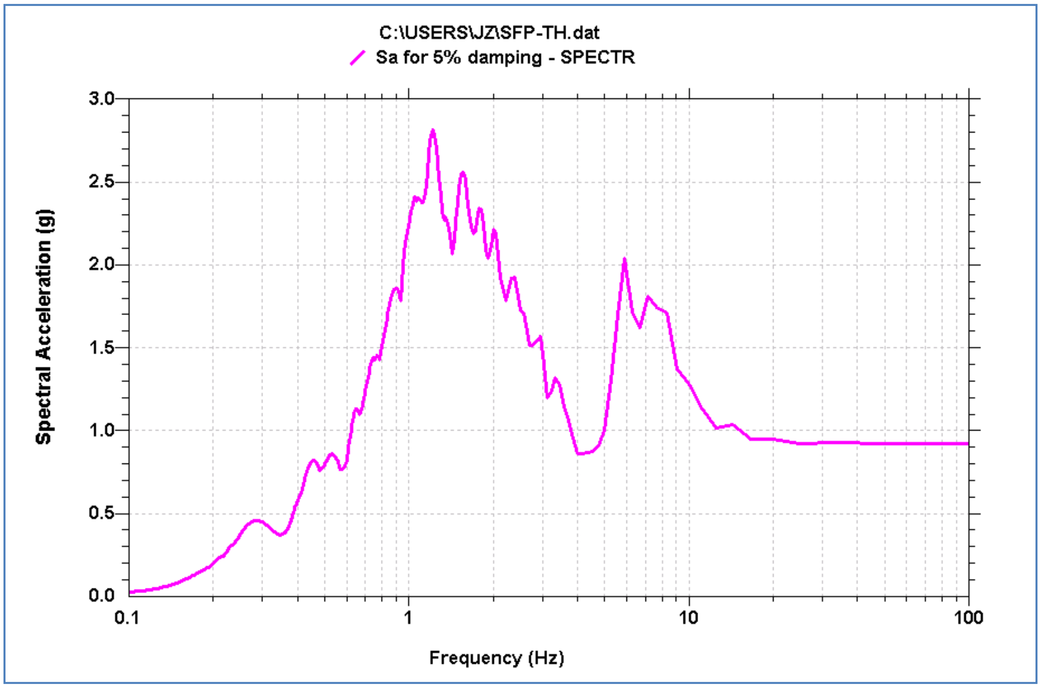
(a)



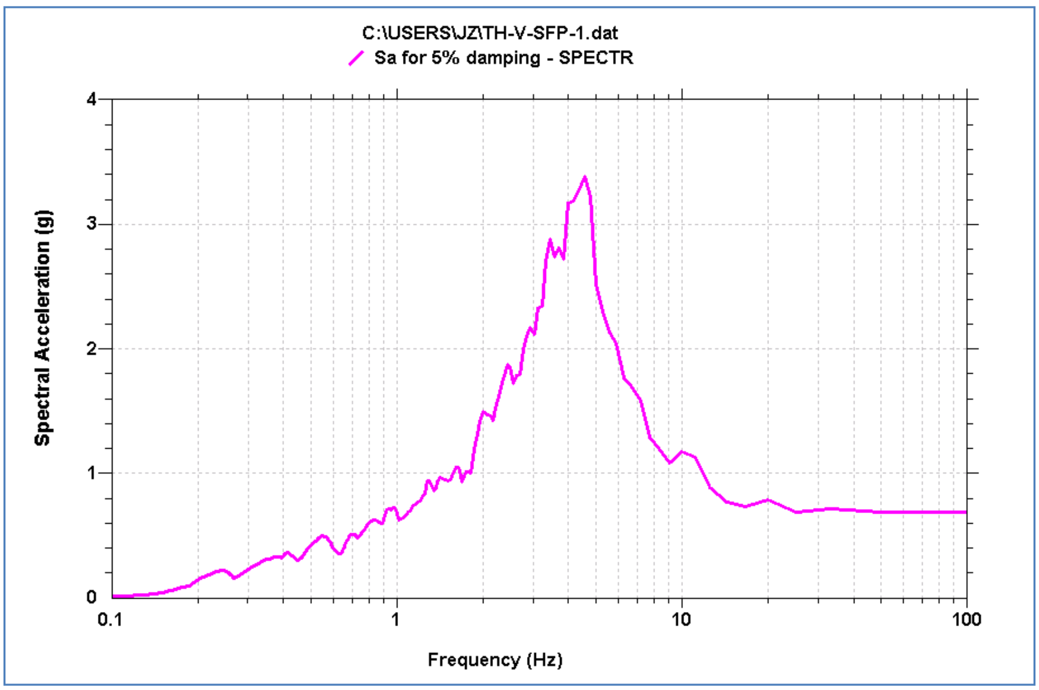
(b)

FIGURE 2.4.1: DESIGN BASIS SPECTRUM AT THE GROUND SURFACE (TOP OF ISFSI PAD) ELEVATION

(a) HORIZONTAL DIRECTION; (b) VERTICAL DIRECTION



(a)



(b)

FIGURE 2.4.2: DESIGN BASIS SPECTRUM AT THE HI-STORM UMAX FOUNDATION SURFACE (BOTTOM OF SFP) ELEVATION

(a) HORIZONTAL DIRECTION; (b) VERTICAL DIRECTION

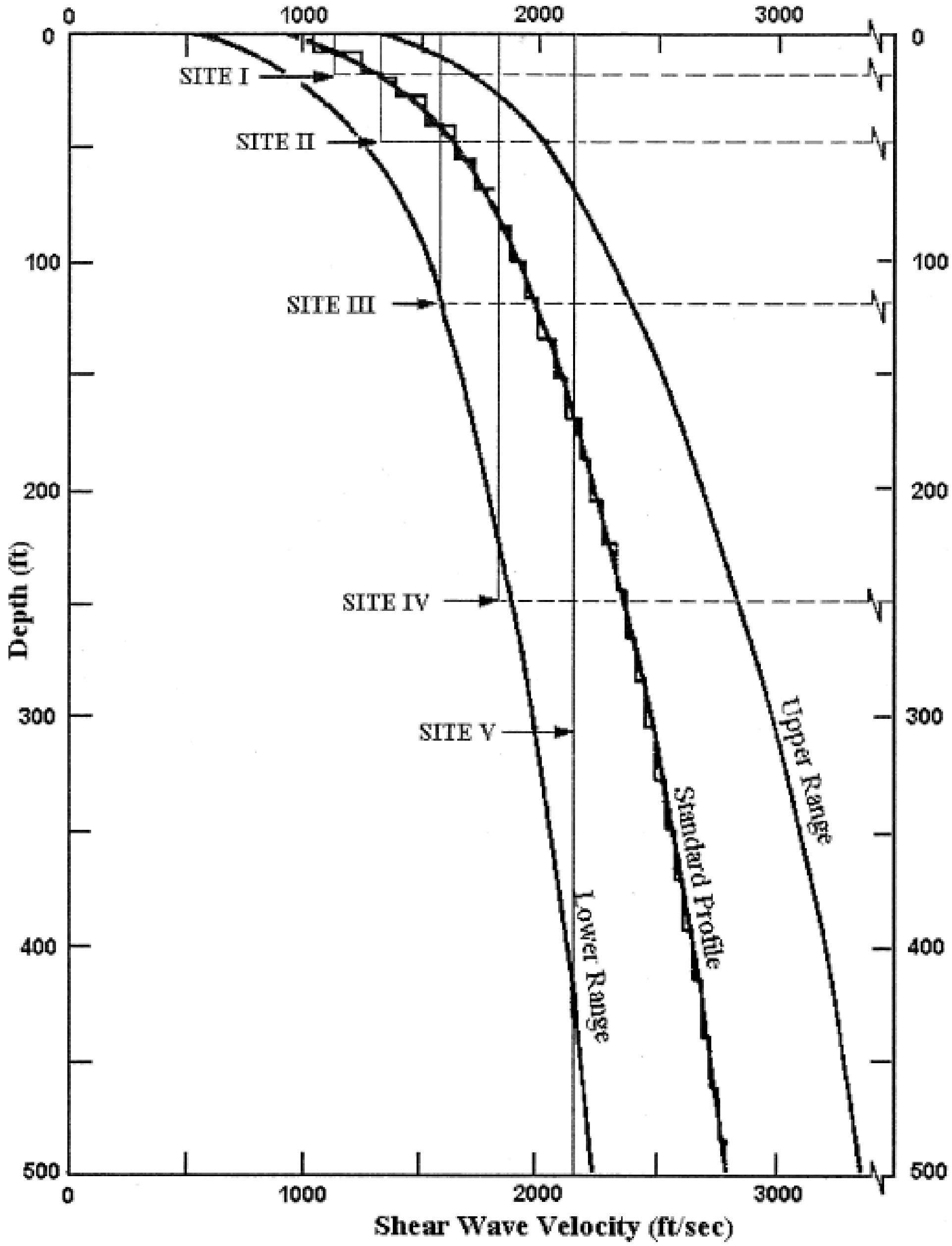


FIGURE 2.4.3: TYPICAL SHEAR WAVE VELOCITY PROFILES FOR NUCLEAR POWER PLANT SITES (REPRODUCED FROM FIGURE I-1 OF [2.4.1]

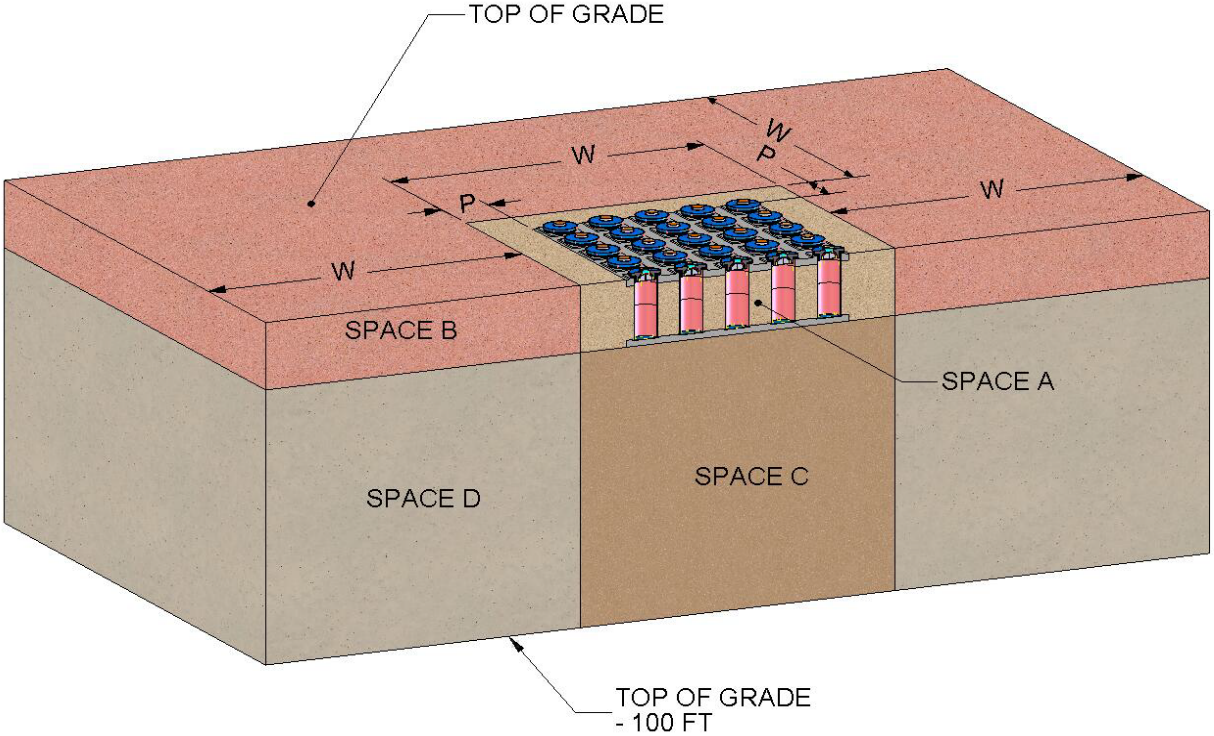


FIGURE 2.4.4: SUB-GRADE AND UNDER-GRADE SPACE NOMENCLATURE

Note 1: Space A is the lateral subgrade space in and around the VVMs which is refilled with CLSM or lean concrete after the construction of the SFP. Space B is the lateral subgrade that extends by the amount W around the ISFSI where W is a representative dimension of the ISFSI determined by site-specific layouts. Space C is the under-grade below the SFP. Space D is the under-grade surrounding Space C. P is the distance to the Enclosure wall.

Note 2: As indicated by the title, this figure is provided to show the nomenclature for the various spaces around a HI-STORM UMAX ISFSI. This figure is not intended to provide specific dimensions.

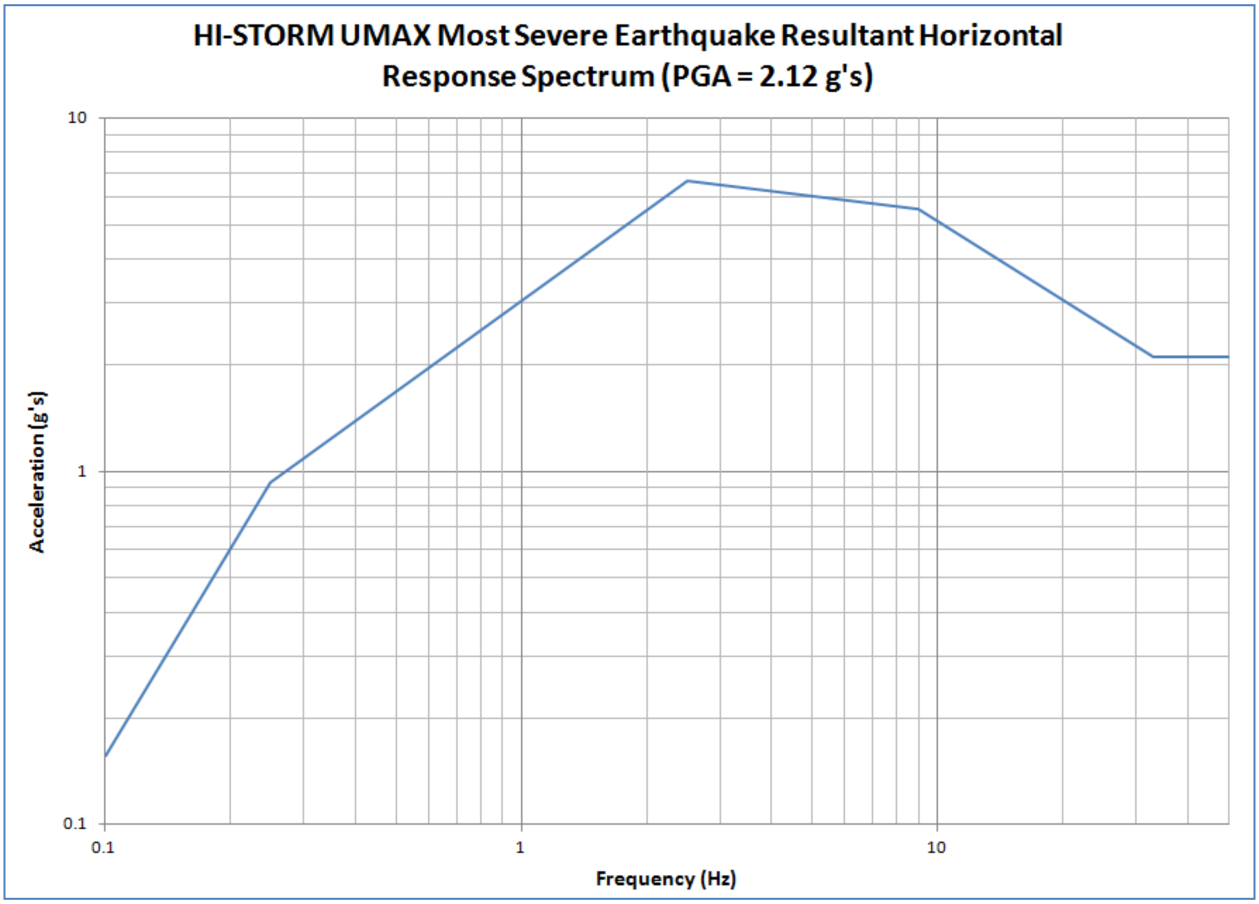


Figure 2.4.5; Horizontal Design Response Spectrum for the Most Severe Earthquake (MSE)

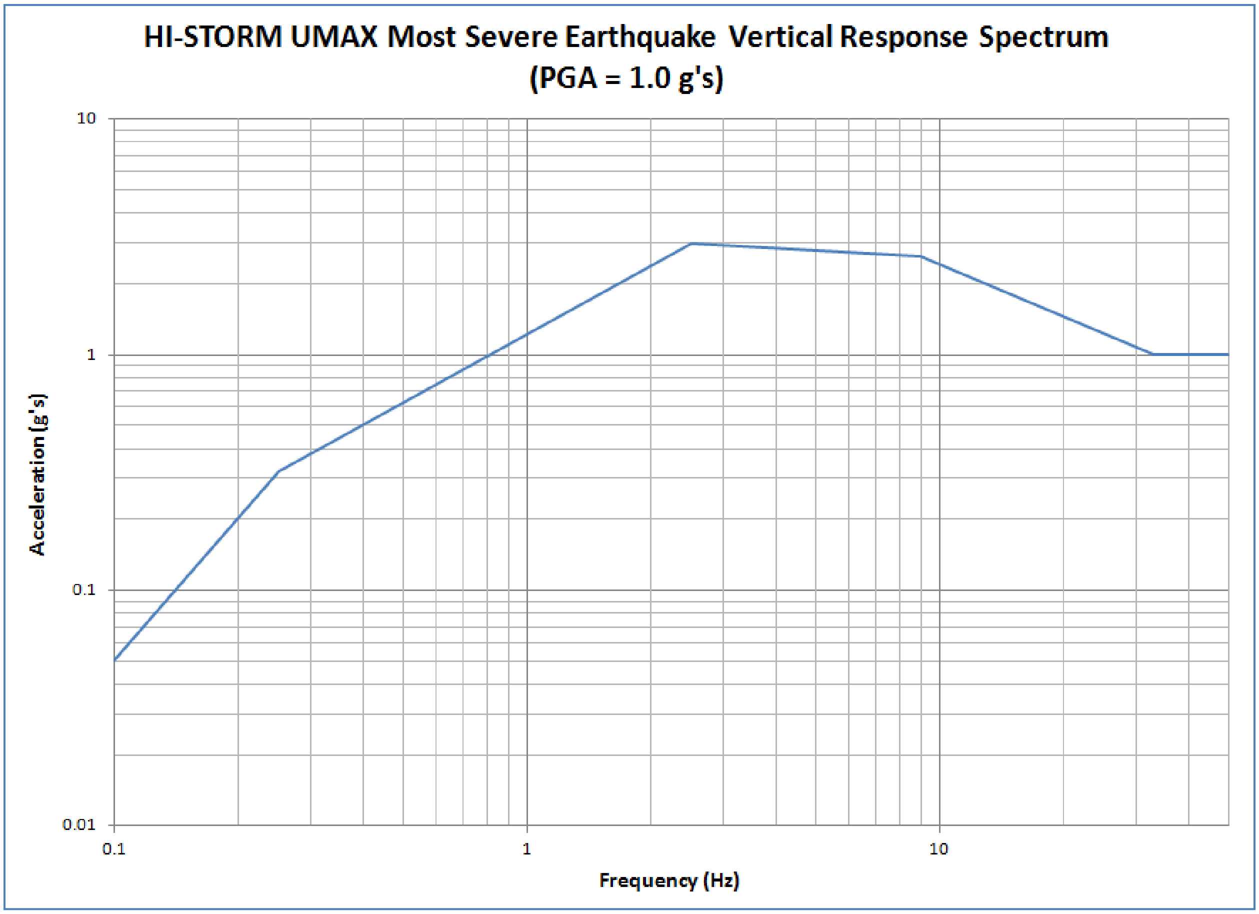


Figure 2.4.6; Vertical Design Response Spectrum for the Most Severe Earthquake (MSE)

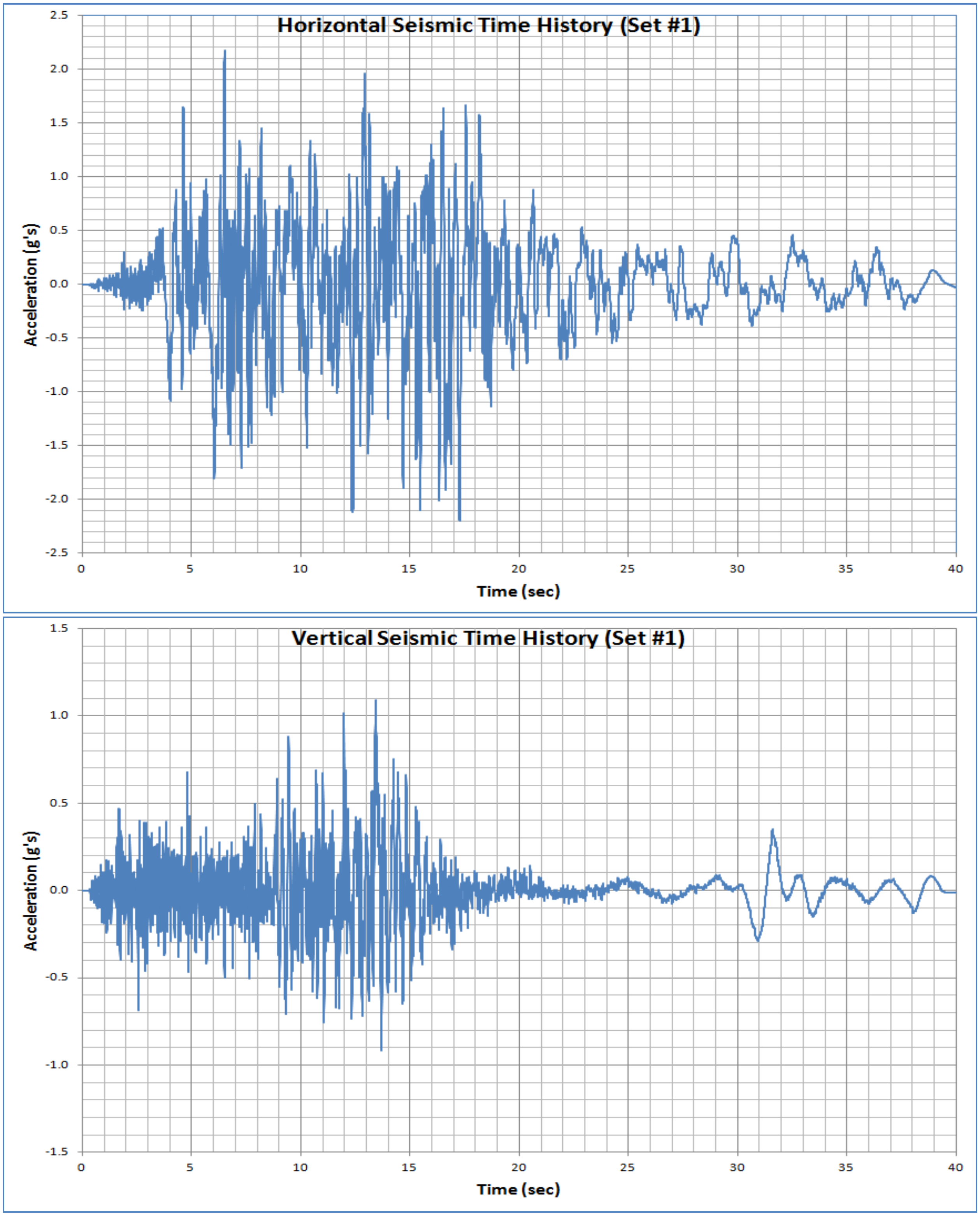
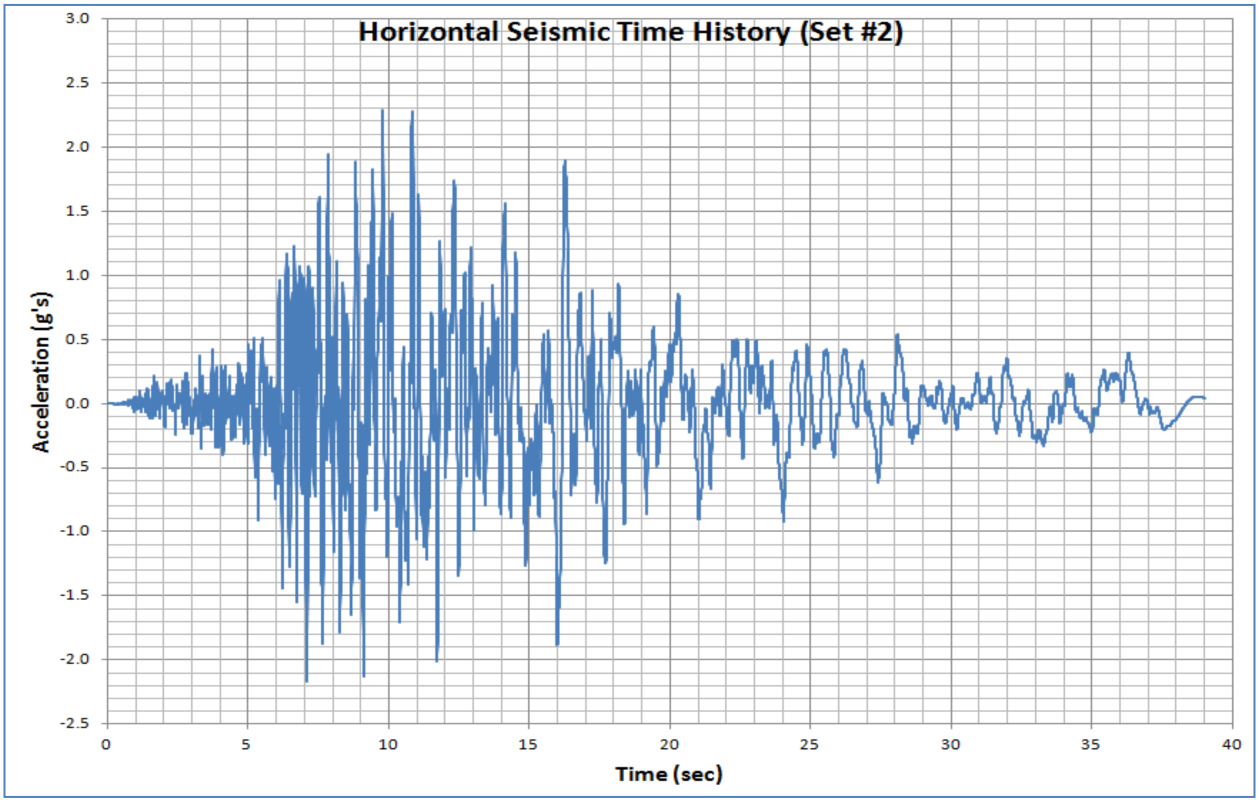


Figure 2.4.7; Accelerograms for Time History Set #1



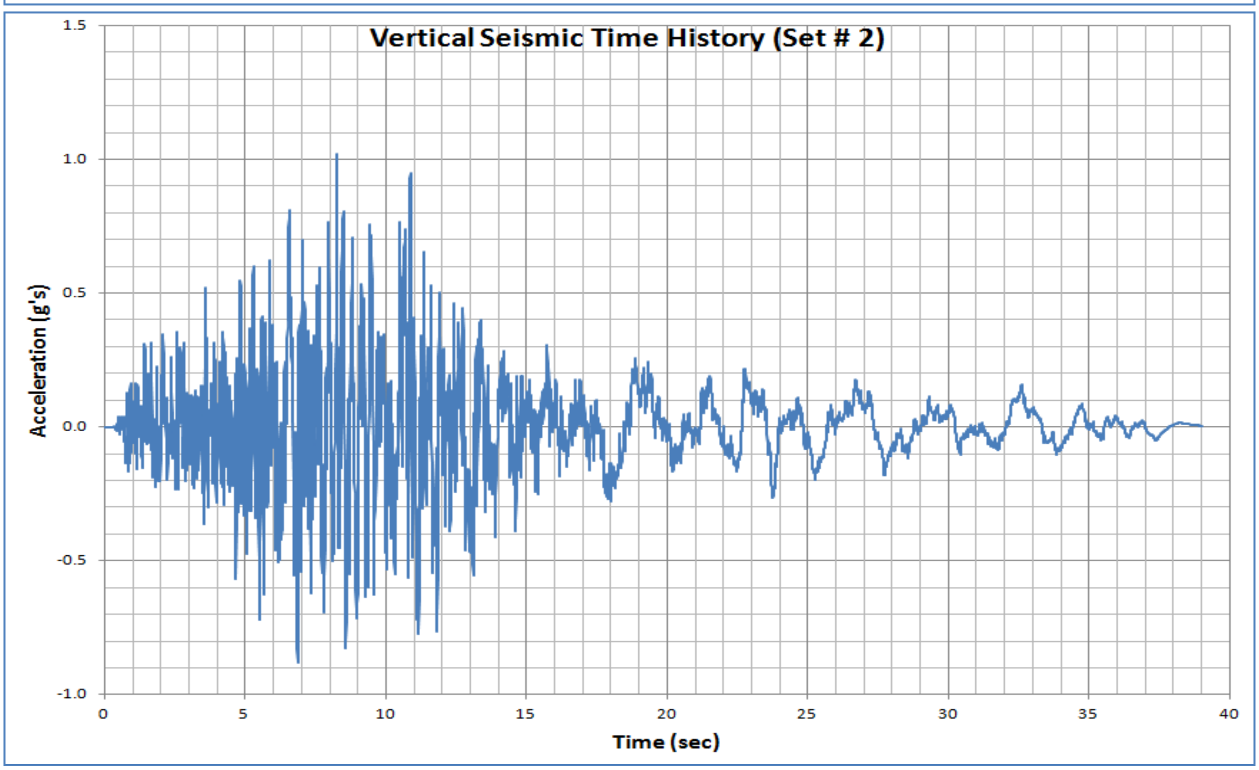


Figure 2.4.8; Accelerograms for Time History Set #2

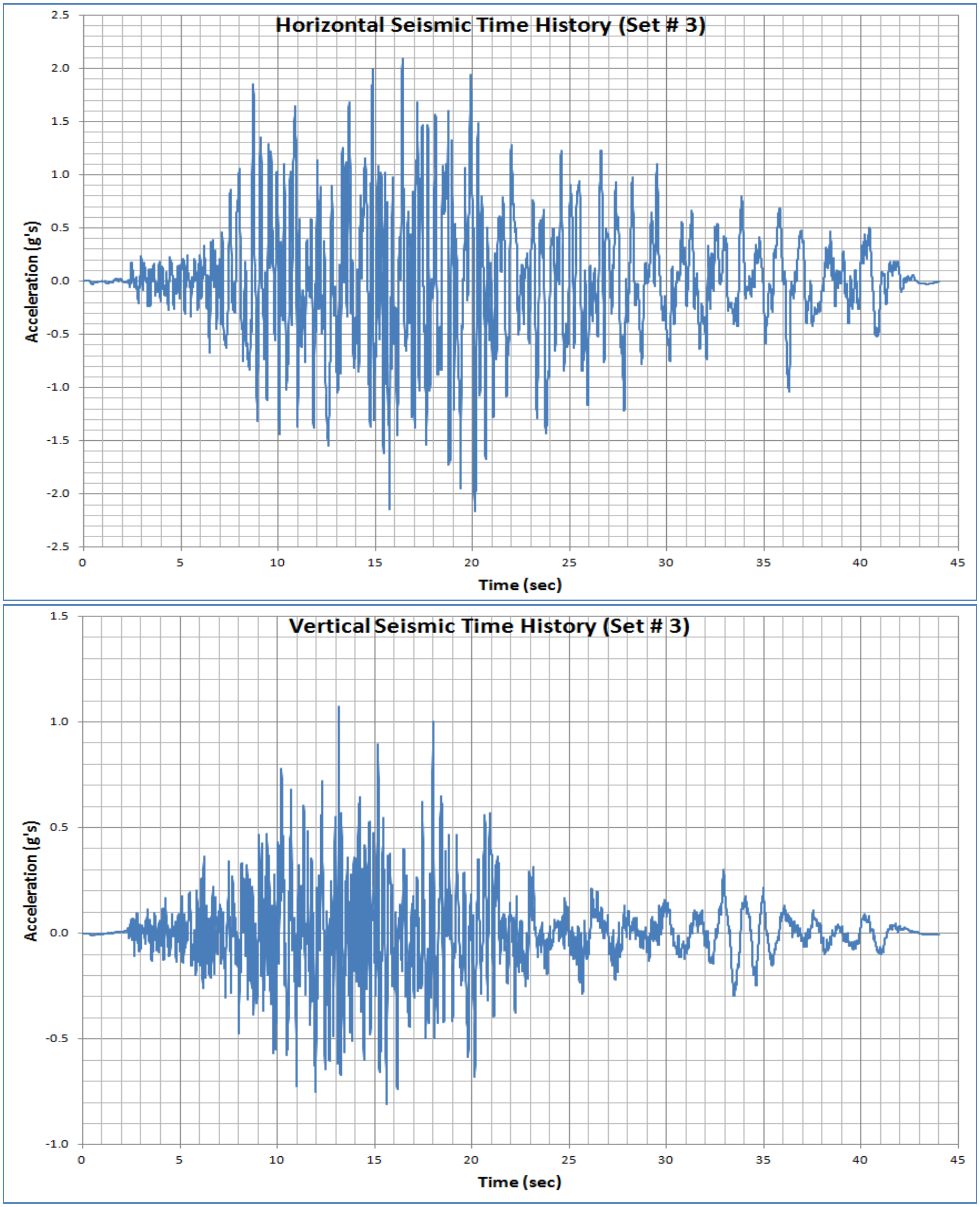
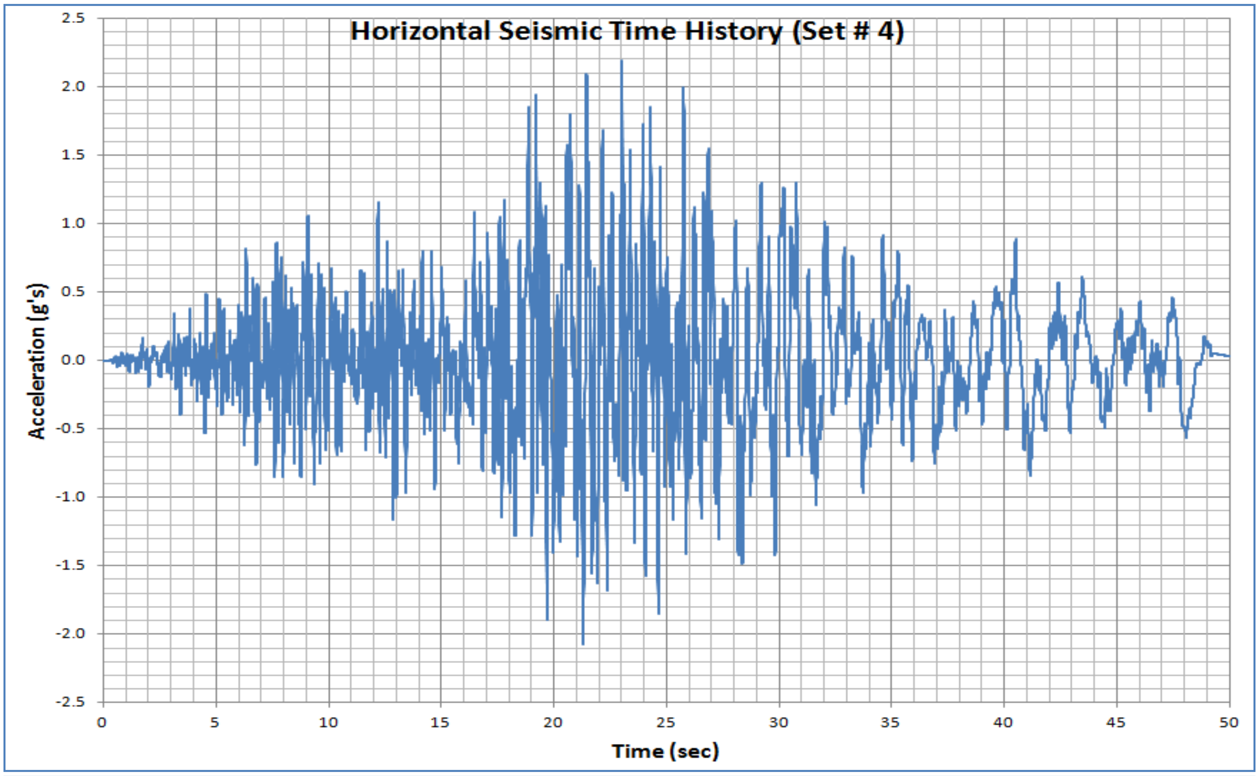


Figure 2.4.9; Accelerograms for Time History Set #3



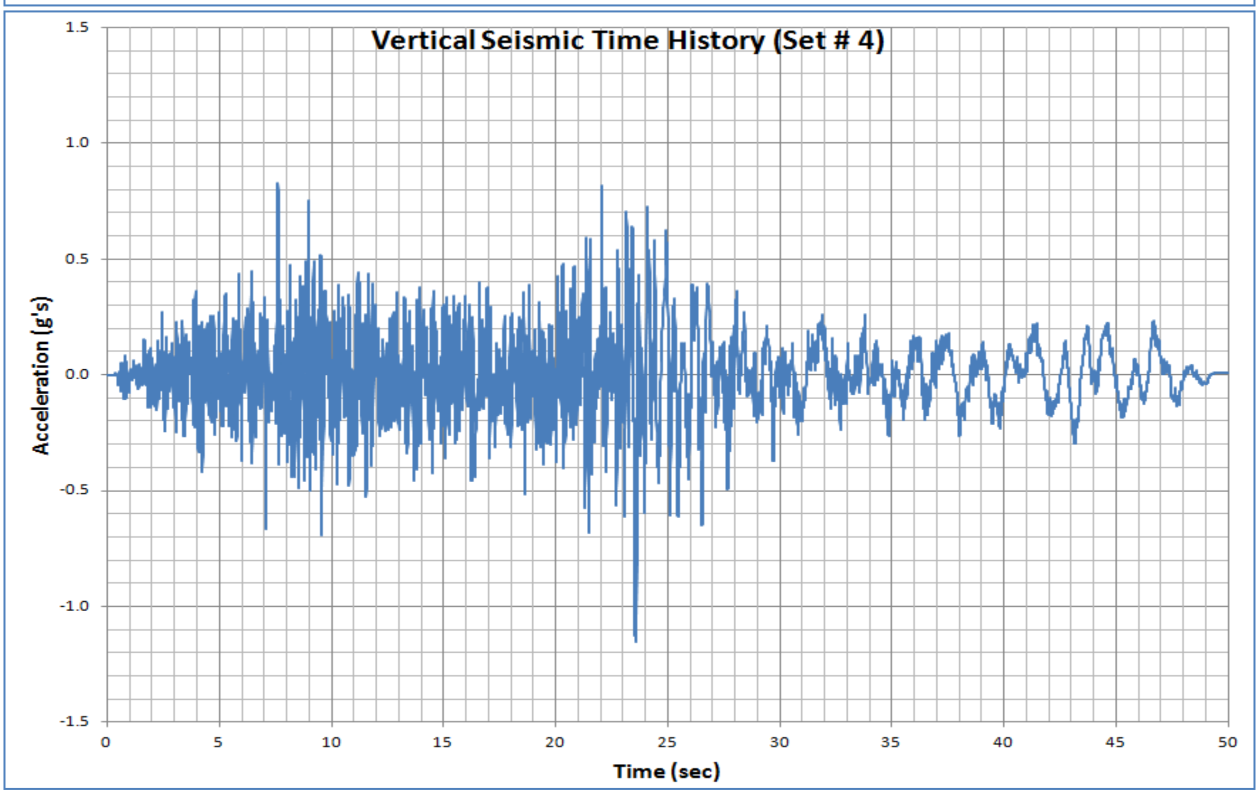


Figure 2.4.10; Accelerograms for Time History Set # 4

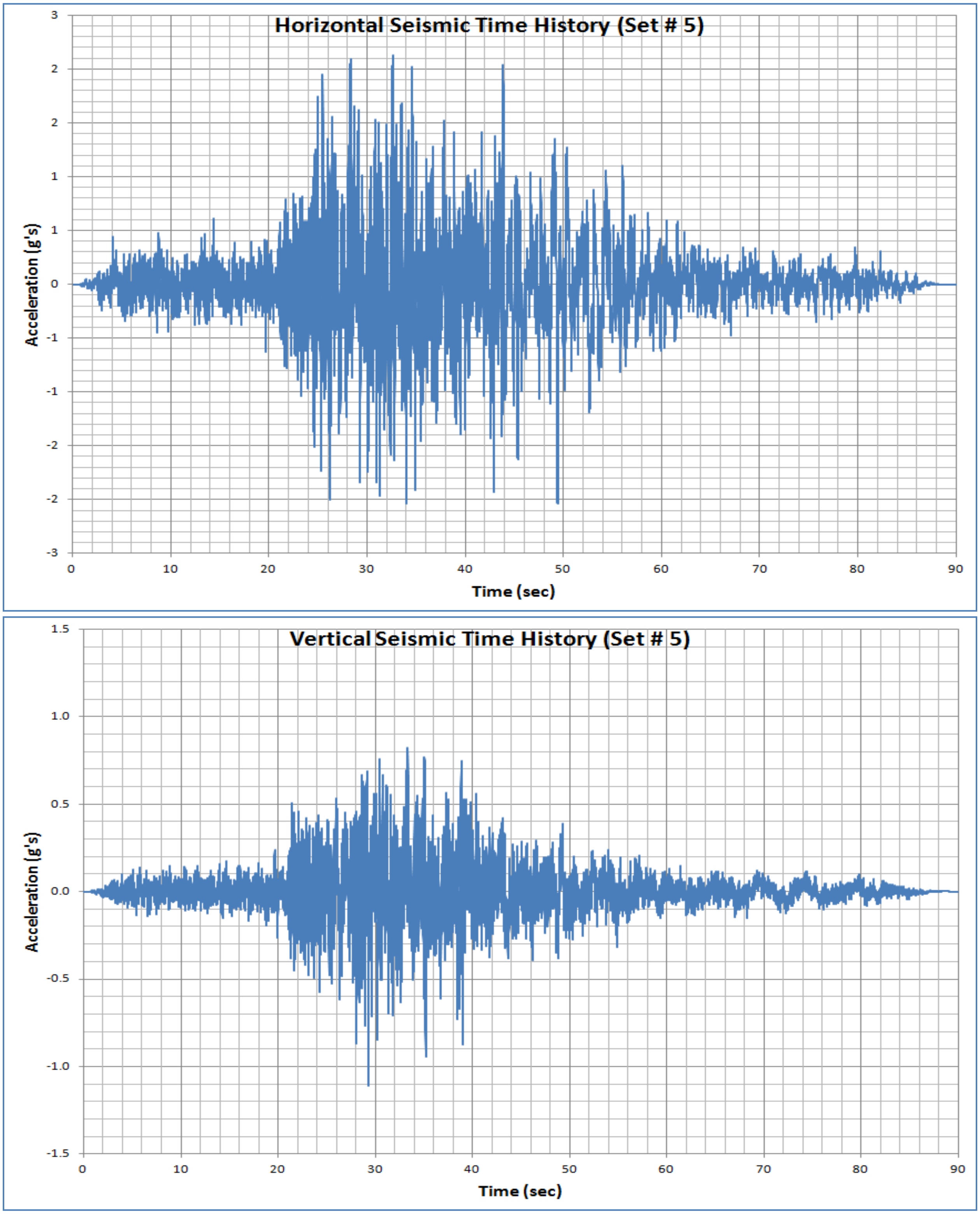


Figure 2.4.11; Accelerograms for Time History Set #5

2.5 THERMALLY SIGNIFICANT LOADS AND ACCEPTANCE CRITERIA

The analyses summarized in this chapter focus on the governing canisters out of the population of MPCs listed in Table 1.2.1. This chapter, however, supports the certification of only MPC-37 and MPC-89 at this time. The analyses reported for smaller canisters are for reference purposes only.

The thermal design and operation of the HI-STORM UMAX System shall meet the intent of the review guidance contained in ISG-11, Revision 3 [2.4.6]. Specifically, the ISG-11 provisions that are explicitly invoked and satisfied are:

a. The thermal acceptance criteria for all commercial spent fuel (CSF) authorized by the USNRC for operation in a commercial reactor are unified into one set of requirements.

b. The maximum value of the calculated temperature for all CSF under long-term normal conditions of storage must remain below 400ºC (752ºF). For short-term operations, including canister drying, helium backfill, and on-site cask transport operations, the fuel cladding temperature must not exceed 400ºC (752ºF) for high burn-up fuel (HBF) and 570ºC (1058ºF) for moderate burn-up fuel.

c. The maximum fuel cladding temperature as a result of an off-normal or accident event must not exceed 570ºC (1058ºF).

d. For HBF, operating restrictions are imposed to limit the maximum temperature excursion during short-term operations to 65ºC (117ºF) and the number of excursions to less than 10.

As stated in Chapter 1, the MPC models designated for storage in the HI-STORM UMAX system are previously certified for use in the HI-STORM FW overpack. For storage in HI-STORM UMAX, additional restriction in the heat load is applicable for certain MPCs and fuel types. Section 2.1 provides the Design Basis Heat Load for storage in HI-STORM UMAX.

The normal condition design temperatures for the materials used in the HI-STORM UMAX system are provided in Table 2.3.7.

Thermally significant loads are characterized by the absence of any significant mechanical loading and are principally applicable to the integrity of the stored fuel. The safety analyses for thermal loadings are contained in Chapter 4. The acceptance criteria for the fuel cladding temperature summarized above, are from ISG 11 Rev 3. The following thermal condition scenarios are applicable to HI-STORM UMAX:

- Normal Ambient Temperature

The HI-STORM UMAX System is analyzed for the same maximum yearly average ambient air temperature as that used for the HI-STORM FW systems (Table 2.3.6) which apply to long-term storage and short-term normal operating conditions (e.g., MPC drying operations and onsite transport operations). Pursuant to NUREG -1536, a certain population of the fuel rods in the MPC is also assumed to have become depressurized due to cladding failure. This normal operating condition temperature bounds all locations in the continental United States.

- Elevated Ambient Air Temperature

The HI-STORM UMAX System must be able to reject the design basis heat load under short-term conditions of elevated ambient air temperature designated as an Off-normal condition pursuant to NUREG-1536.

- Partial Blockage of Inlet Air Plenum

Pursuant to NUREG-1536, 50% of the inlet ducts are assumed to be blocked under an off normal storage condition.

The HI-STORM UMAX System must withstand 50% blockage of the inlet air flow plenum without exceeding allowable temperature and pressure limits specified for the off-normal condition.

- 100% Blockage of Air Inlets by Debris

The HI-STORM UMAX is assumed to be subject to a complete blockage of the inlet ducts. This is assumed to occur as a postulated accident event. Chapter 4 contains the appropriate thermal analysis which serves to establish the ISFSI surveillance program in the CoC for the system.

- 100% Fuel Rod Rupture

The 100% rod rupture event is a *non-mechanistic* postulate intended to define a bounding scenario of rise in the MPC internal pressure. The HI-STORM UMAX System must withstand loads due to 100% fuel rod rupture. For conservatism, 100% of the fuel rods are assumed to rupture with 100% of the fill gas and 30% of the significant radioactive gases (e.g., H^3, Kr, and Xe) released in accordance with NUREG-1536. All of the fill gas contained in non-fuel hardware, such as burnable poison rod assemblies (BPRAs), is also assumed to be released concomitantly.

The acceptance criterion for this loading event is that the accident condition MPC Design Pressure (Table 2.3.5) is not exceeded.

- Wind

Wind is a common environmental condition which is characterized by varying magnitude and direction at all terrestrial locations. Because wind is an ever changing condition, steady state conditions cannot be expected to be reached in the HI-STORM UMAX storage system which has a substantial thermal inertia. However, it is necessary to ensure that the heat rejection function of the system is not significantly impaired by wind. To make this determination, the thermal performance of the system is quantified under a sustained wind (assumed to be of sufficient duration so that steady state conditions are reached) at typical velocities of in the range of 0 to 10 MPH. Sustained wind in a fixed direction for an extended period is not a plausible environmental occurrence event. Because in NUREG-1536, this condition is not specified as one requiring evaluation, this evaluation performed in this FSAR, exceeds the scope specified in NUREG-1536.

- Burial Under Debris

The HI-STORM UMAX vent screens are engineered to prevent accumulation of dust and debris. Siting of the ISFSI pad shall ensure that the storage location is not located over shifting soil.

However, it may be possible for wind borne debris or debris from a collapsing structure in the vicinity of the ISFSI to block the inlet plenum. If burial of an extensive portion of the VVM is a credible event for an ISFSI, then a thermal analysis to analyze the effect of such an accident condition shall be performed for the site using the analysis methodology presented in Chapter 4. The duration of the burial-under-debris scenario will be based on the ISFSI owner's emergency preparedness program. The following acceptance criteria apply to the burial-under-debris accident event:

The fuel cladding temperature shall not exceed the ISG-11, Revision 3 [2.4.6] temperature limits.

The internal pressure in the MPC cavity shall not exceed the accident condition design pressure limit in Table 2.3.5.

The burial-under-debris analysis will be performed if applicable, for the site-specific conditions and heat loads.

- Extreme Environmental Temperature

The HI-STORM UMAX System must withstand extreme environmental temperatures. The extreme accident level temperature is specified in Table 2.3.6. The extreme accident level temperature is assumed to occur with steady-state insolation. This temperature is assumed to persist for a sufficient duration to allow the system to reach steady-state temperatures. As is standard for ventilated systems, extreme environmental temperature is a 3-day average for the ISFSI site.

- Design Basis Fire

The possibility of a fire accident near an ISFSI site is considered to be extremely remote due to the absence of significant combustible materials. The only credible concern is related to a transport vehicle fuel tank fire engulfing the loaded HI-STORM UMAX VVM or loaded HI-TRAC VW transfer cask while it is being moved to the ISFSI.

The HI-TRAC and HI-STORM UMAX VVM must withstand temperatures due to a fire event. The HI-STORM UMAX fire accidents for storage are conservatively postulated to be the result of the spillage and ignition of 50 gallons of combustible transporter fuel (identical to the HI-STORM 100 docket). The HI-STORM UMAX lid external surfaces are considered to receive an incident radiation and forced convection heat flux from the fire. The temperature of fire is assumed to be 1475° F in accordance with 10CFR71.73.

The HI-STORM UMAX System must withstand fire accident without exceeding allowable temperature and pressure limits specified for the accident condition.

- Flood

A potentially severe flood event could happen during the storage period. In that event, the water could enter the inlet ducts and block portion or the entire cooling air flow passageway at the bottom of the cavity, which reduces the air flow ventilating through VVM and causes an elevation of the fuel cladding temperature and system component temperatures.

Chapter 4 contains the appropriate thermal analysis which serves to establish the ISFSI surveillance program in the CoC for the system.

- Jacket Water Loss

The fuel cladding and MPC boundary integrity should be evaluated under a postulated (non-mechanistic) complete loss of water from the HI-TRAC VW water jacket. The HI-TRAC VW must withstand the jacket water loss accident without exceeding allowable temperature and pressure limits specified for the accident condition.

2.5.1 The Forced Helium Dehydrator

The Forced Helium Dehydrator (FHD) is a HI-STORM system ancillary used to remove the remaining moisture in the MPC cavity after all of the water that can practically be removed through the drain line using a hydraulic pump or an inert gas has been expelled in the water blow-down operation. The FHD system is required to be used for MPCs loaded with one or more high burnup fuel assemblies and generating greater than threshold heat loads defined in Chapter 4. The FHD method of moisture removal is optional for all other MPCs.

Expelling the water from the MPC using a conventional pump or a water displacement method using inert gas would remove practically all of the contained water except for the small quantity remaining on the MPC baseplate below the bottom of the drain line and an even smaller adherent amount wetting the internal surfaces. A skid-mounted, closed loop dehydration system will be used to remove the residual water from the MPC such that the partial pressure of the trace quantity of water vapor in the MPC cavity gas is brought down to ≤ 3 torr. The FHD system, engineered for this purpose, utilizes helium gas as the working substance. In comparison to the classical vacuum drying process, the FHD maintains the fuel cladding at a relatively low temperature (substantially below the permissible cladding temperature in ISG-11 Rev 3) which insures that the pressure inside the cladding and hence its hoop stress remains at a moderate level. The design features of the FHD are described in an array of USPTO-issued patents available in the open literature as follows:

- Patent Number 7,096,600B2, Forced Helium Dehydrator, dated August 29, 2006
- Patent Number 7,210,247B2, Forced Gas Flow Canister Dehydration, dated May 1, 2007
- Patent Number 7,707,741B2, Dew Point Temperature Based Canister Dehydration, dated May 4, 2010
- Patent Number 8,067,659B2, Method of Removing Radioactive Materials from Submerged State and/or Preparing Spent Nuclear Fuel for Dry Storage, dated November 29, 2011
- Patent Number 8,266,823B2, Method and Apparatus for Dehydrating High Level Waste Based on Dew Point Temperature Measurements, dated September 18, 2012

2.5.1.1 FHD Design Criteria

The Design of the FHD has been standardized for all HI-STORM/HI-STAR MPCs using the design criteria set forth in the following which have remained unchanged since their first adoption in the HI-STORM 100 FSAR over a decade ago. These design criteria are intended to ensure that design and operation of the FHD system will remove bulk moisture and lower the partial pressure of the residual vapor in the MPC cavity to ≤ 3 torr if the circulating gas reaches

the specified temperature or dew point value and duration criteria. The FHD system is designed to ensure that during normal operation (i.e., excluding startup and shutdown ramps) the following criteria are met:

i. The temperature of helium gas in the MPC shall be at least 15ºF higher than the saturation temperature at coincident pressure. The operating pressure shall be selected to maximize the mass flow rate within the constraints of the pressure rating of the associated pressure parts.

ii. The recirculation rate of helium shall be sufficiently high (minimum hourly throughput equal to ten times the nominal helium mass backfilled into the MPC for fuel storage operations) so as to produce a turbulated flow regime in the MPC cavity.

iii. The partial pressure of the water vapor in the MPC cavity will not exceed 3 torr. This limit will be met if the gas temperature at the de-moisturizer outlet is verified by measurement to remain ≤ 21ºF for ≥ 30 minutes or if the dew point of the gas exiting the MPC is verified by measurement to remain ≤ 22.9ºF for ≥ 30 minutes.

The design of the FHD ancillary and the thermal- hydraulic simulations to insure that it would meet the above design criteria were carried out at the time of its introduction in the HI-STORM 100 system in 2001. A Holtec proprietary report titled “Forced Helium Dehydrator Sourcebook” Holtec Report HI-2022966 documents the design and confirmatory analyses on the FHD. As required by the HI-STORM 100 FSAR (in its Appendix 2.B), the first FHD manufactured and deployed in MPC drying was subjected to acceptance testing. Since then, over 20 nuclear units have opted for the FHD method of drying. Like all ancillaries, the entire body of information on the design, manufacturing and testing of the FHDs is maintained for archival reference in Holtec’s quality assurance system. The FHD has been designated not-important-to-safety (NITS) because its malfunction cannot precipitate a rise in the stored fuel's reactivity, cause increase in the cladding temperature above design limits, cause a significantly elevated dose rate to the crew or lead to release of radioactive matter to the environment. Instrumentation used to ensure that the licensed conditions for operation are met is calibrated by vendors who are on the Holtec Approved Vendors List for Important-to-Safety and Safety Related equipment.

2.6 MATERIALS, CODES, STANDARDS, AND PRACTICES TO ENSURE REGULATORY COMPLIANCE

There is no U.S. or international code that is sufficiently comprehensive to provide a completely prescriptive set of requirements for the design, manufacturing, and structural qualification of the VVM. The various sections of the ASME Codes, however, contain a broad range of specifications that can be assembled to provide a complete set of requirements for the design, analysis, shop manufacturing, and field erection of the VVMs. The portions of the ASME Codes that are invoked for the various elements of the VVM design, analysis, and manufacturing activities are summarized in Table 2.6.1.

The ASME Boiler and Pressure Vessel Code (ASME Code) Section III, Subsection NF Class 3 [2.6.1], is the applicable code to determine stress limits for the metallic structural components of the VVM when required by the acceptance criteria listed in Tables 2.4.1 and 2.4.2. The permitted material types for long-term use are listed in Table 2.6.2. Manufacturing requirements are set down in licensing and design drawings.

Section III Subsection NB of the ASME Boiler and Pressure Vessel Code [2.6.8], is the governing code for the structural design of the MPC. The alternatives to the ASME Code, Section III Subsection NB, applicable to the MPC in Docket Nos. 72-1032 are also applicable to the MPC in the HI-STORM UMAX System, as documented in Table 2.6.5.

The stress limits of ASME Section III Subsection NF [2.6.10] are applied to the HI-TRAC structural parts where the applicable loading is designated as a code service condition.

The fuel basket, made of Metamic-HT, is subject to the requirements in Metamic-HT Sourcebook [1.2.4] and is designed to a specific (lateral) deformation limit of its walls under accident conditions of loading (credible and non-mechanistic) (see Table 2.2.11 of HI-STORM FW FSAR). The basis for the lateral deflection limit in the active fuel region, θ, is provided in [2.6.9].

ACI-318(2005) [2.6.2] is the applicable reference code to establish applicable limits on unreinforced concrete (in the Closure Lid), which is subject to secondary structural loadings. Chapter 8 contains the design, construction, and testing criteria applicable to the plain concrete in the VVM's Closure Lid. The load combinations applicable to the ISFSI pad and the Enclosure Wall, pursuant to ACI-318(05) are summarized in Table 2.4.3 Applicable sections of ACI-318(2005) should be used in the design of the ISFSI pad and the SFP. Reference [2.6.5] should be used as a secondary guidance document in designing the ISFSI pad. The applicable provisions of [2.6.6] are invoked as an aid in defining the subgrade space that should be modeled in the soil/structure interaction simulation.

The selection of the ISFSI site shall be made with due consideration of the potential of liquefaction. The host plant's criteria with respect to liquefaction for siting the Part 50 structures shall be used in evaluating the suitability of a candidate ISFSI location.

As mandated by 10CFR72.24(c)(3) and §72.44(d), Holtec International's quality assurance program requires all constituent parts of an SSC subject to NRC's certification under 10CFR72 to be assigned an ITS category appropriate to its function in the control and confinement of

radiation. The ITS designations for the constituent parts of the HI-STORM UMAX VVM, using the guidelines of NUREG-CR/6407 [2.6.4], are provided in the Licensing drawing package.

The aggregate of the citations from the codes, standards, and generally recognized industry publications invoked in this FSAR, supplemented by the commitments in Holtec's quality assurance procedures, provide the necessary technical framework to ensure that the as-installed VVM s would meet the intent of §72.24(c), §72.120(a) and §72.236(b). As required by Holtec's QA Program docketed with the NRC (Docket Number 71-0784), all operations on ITS components must be performed under QA validated written procedures and specifications that are in compliance with the governing citations of codes, standards, and practices set down in this FSAR. For activities that may be performed by others, such as site construction work to install the VVM, Holtec International requires that all activities be formalized in procedures and subject to the CoC holder's as well as the ISFSI owner's review and approval.

An ITS designation is also applied to the interfacing structures (such as the SFP), which requires that all quality assurance measures set down in Holtec's Quality Assurance Procedure Manual be complied with by the entity performing the site construction work. In this manner, the compliance of the as-built VVMs with its engineered safety margins under all design basis scenarios of loading is assured.

Table 2.6.1

REFERENCE ASME CODE PARAGRAPHS FOR VVM PRIMARY LOAD BEARING PARTS

	Item	Code Paragraph [2.6.1]	Explanation and Applicability
1.	Definition of primary and secondary members	NF-1215	-
2.	Jurisdictional boundary	NF-1133	The VVM's jurisdictional boundary is defined by the bottom surface of the SFP, the top surface of the ISFSI pad and the SES side surfaces.
3.	Certification of material(structural)	NF-2130(b) and (c)	Materials shall be certified to the applicable Section II of the ASME Code or equivalent ASTM Specification.
4.	Heat treatment of material	NF-2170 and NF-2180	-
5.	Storage of welding material	NF-2400	-
6.	Welding procedure	Section IX	-
7.	Welding material	Section II	-
8.	Loading conditions	NF-3111	-
9.	Allowable stress values	NF-3112.3	-
10.	Rolling and sliding supports	NF-3424	-
11.	Differential thermal expansion	NF-3127	-
12.	Stress analysis	NF-3143 NF-3380 NF-3522 NF-3523	Provisions for stress analysis for Class 3 plate and shell supports and for linear supports are applicable for Closure Lid and Container Shell, respectively.
13.	Cutting of plate stock	NF-4211 NF-4211.1	-
14.	Forming	NF-4212	-
15.	Forming tolerance	NF-4221	Applies to the Container Shell
16.	Fitting and Aligning Tack Welds	NF-4231 NF-4231.1	-
17.	Alignment	NF-4232	-
18.	Storage of Welding Materials	NF-4411	-
19.	Cleanliness of Weld Surfaces	NF-4412	Applies to structural and non-structural

Table 2.6.1

REFERENCE ASME CODE PARAGRAPHS FOR VVM PRIMARY LOAD BEARING PARTS

	Item	**Code Paragraph [2.6.1]**	**Explanation and Applicability**
			welds
20.	Backing Strips, Peening	NF-4421 NF-4422	Applies to structural and non-structural welds
21.	Pre-heating and Interpass Temperature	NF-4611 NF-4612 NF-4613	Applies to structural and non-structural welds
22.	Non-Destructive Examination	NF-5360	Invokes Section V
23.	NDE Personnel Certification	NF-5522 NF-5523 NF-5530	-

Table 2.6.2

PRINCIPAL MATERIALS, THEIR FUNCTION & ITS CATEGORIES FOR VVM

Item	Primary Function	Part	ITS Category	Material
1.	Shielding	Closure Lid Concrete	C	Shielding Concrete
2.	Shielding	Closure Lid Steel	C	ASTM A516, Gr. 70, A515 Gr. 70 or equivalent
3.	Structural	Container Shell, Bottom Plate and Container Flange	C	ASTM A516, Gr. 70, A515 Gr. 70 or equivalent
4.	Thermal	Insulation	C	Commercial
5.	Thermal	Inlet/Outlet Vent Screens and associated hardware	NITS	Carbon steel, stainless steel, aluminum, polymeric fabric or commercial
6.	Thermal	Outlet Vent Cover and associated hardware	NITS	Carbon steel, stainless steel, aluminum or commercial
7.	Rain Protection	Vent Flue	NITS	Aluminum
8.	Non-Structural (emulates a shim)	MPC Bearing Pad	NITS	Carbon Steel (with stainless steel liners**)**
9.	Shielding and Physical Protection	ISFSI Pad	C	Reinforced Concrete Per ACI-318 (2005)
10.	Shielding and Physical Protection	ISFSI Pad Subgrade Surrounding the VVMs	C	Self-hardening Engineered Subgrade (SES)
11.	Structural Support	Support Foundation Pad (SFP)	C	Reinforced Concrete per ACI-318 (2005)
12.	Barrier against water ingress in the ISFSI space and/or as the “form” for SES placement	Enclosure wall (optional)	NITS	Concrete or another form of moisture resistant barrier

Table 2.6.3

CRITICAL CHARACTERISTICS OF MATERIALS REQUIRED FOR SAFETY EVALUATION OF STORAGE AND TRANSPORT SYSTEMS

Item	**Property**	**Type (Note 1)**	**Purpose**	**Bounding Acceptable Limit**
1.	Minimum Yield Strength	S	To ensure adequate elastic strength for normal service conditions	Min.
2.	Minimum Tensile Strength	S	To ensure material integrity under accident conditions	Min.
3.	Young's Modulus	S	For input in structural analysis model	Min.
4.	Minimum elongation of $\delta_{min.}$, %	S	To ensure adequate material ductility	Min.
5.	Impact Resistance at ambient conditions	S	To ensure protection against crack propagation	Min.
6.	Maximum allowable creep rate	S	To prevent excessive deformation under steady state loading at elevated temperatures	Max.
7.	Insulation Thermal conductivity (maximum averaged value in the range of ambient to maximum service temperature, t_{max})	T	To reduce the transmission of decay heat from the MPC to the down-coming cool air in the annular gap between divider shell and container shell	Max.
8.	Thermal conductivities for basket (minimum averaged value in the range of ambient to maximum service temperature, t_{max})	T	To ensure that the basket will conduct heat at the rate assumed in its thermal model	Min.
9.	Minimum Emissivity	T	To ensure that the thermal calculations are performed conservatively	Min.
10.	Specific Gravity	S (and R)	To compute weight of the component (and shielding effectiveness)	Max. (and Min.)
11.	Thermal Expansion Coefficient	T (and S)	To compute the change in basket dimension due to temperature (and thermal stresses)	Min. (and Max.)
12.	Boron-10 Content	R	To control reactivity	Min.

Note 1: **Technical Area of Applicability**

S - Those needed to ensure structural compliance

T - Those needed to ensure compliance with thermal (temperature limits)

R - Those needed to ensure radiation (criticality and shielding) compliance

Table 2.6.4

CRITICAL CHARACTERISTICS OF EQUIVALENT MATERIALS USED IN THE VVM COMPONENTS

Designated Material	**Item**	**Critical Characteristic**
ASTM A515 or A516, Gr. 70	Yield Strength	Yield strength vs. Temperature data must exceed values from appropriate tables for 515/516 Gr.70 materials in ASME Code, Section II, Part D at all applicable temperatures.
	Ultimate Strength	Ultimate strength vs. Temperature data must exceed values from appropriate tables for 515/516 Gr.70 materials in ASME Code, Section II, Part D at all applicable temperatures.
	Elongation	Elongation must equal or exceed value(s) for 515/516 Gr. 70
	Charpy Impact	Values that measure resistance to impact must equal or exceed corresponding values for 515/516 Gr. 70.

Table 2.6.5

LIST OF ASME CODE ALTERNATIVES FOR MULTI-PURPOSE CANISTERS (MPCS)

MPC Enclosure Vessel	Subsection NCA	General Requirements. Requires preparation of a Design Specification, Design Report, Overpressure Protection Report, Certification of Construction Report, Data Report, and other administrative controls for an ASME Code stamped vessel.	Because the MPC is not an ASME Code stamped vessel, none of the specifications, reports, certificates, or other general requirements specified by NCA are required. In lieu of a Design Specification and Design Report, the HI-STORM FSAR includes the design criteria, service conditions, and load combinations for the design and operation of the MPCs as well as the results of the stress analyses to demonstrate that applicable Code stress limits are met. Additionally, the fabricator is not required to have an ASME-certified QA program. All important-to-safety activities are governed by the NRC-approved Holtec QA program. Because the cask components are not certified to the Code, the terms “Certificate Holder” and “Inspector” are not germane to the manufacturing of NRC-certified cask components. To eliminate ambiguity, the responsibilities assigned to the Certificate Holder in the Code, as applicable, shall be interpreted to apply to the NRC Certificate of Compliance (CoC) holder (and by extension, to the component fabricator) if the requirement must be fulfilled. The Code term “Inspector” means the QA/QC personnel of the CoC holder and its vendors assigned to oversee and inspect the manufacturing process.

Table 2.6.5

LIST OF ASME CODE ALTERNATIVES FOR MULTI-PURPOSE CANISTERS (MPCS)

MPC Enclosure Vessel	NB-1100	Statement of requirements for Code stamping of components.	MPC Enclosure Vessel is designed and will be fabricated in accordance with ASME Code, Section III, Subsection NB to the maximum practical extent, but Code stamping is not required.
MPC basket supports and lift lugs	NB-1130	NB-1132.2(d) requires that the first connecting weld of a non-pressure retaining structural attachment to a component shall be considered part of the component unless the weld is more than 2t from the pressure retaining portion of the component, where t is the nominal thickness of the pressure retaining material. NB-1132.2(e) requires that the first connecting weld of a welded nonstructural attachment to a component shall conform to NB-4430 if the connecting weld is within 2t from the pressure retaining portion of the component.	The lugs that are used exclusively for lifting an empty MPC are welded to the inside of the pressure-retaining MPC shell, but are not designed in accordance with Subsection NB. The lug-to-Enclosure Vessel Weld is required to meet the stress limits of Reg. Guide 3.61 in lieu of Subsection NB of the Code.
MPC Enclosure Vessel	NB-2000	Requires materials to be supplied by ASME-approved material supplier.	Materials will be supplied by Holtec approved suppliers with Certified Material Test Reports (CMTRs) in accordance with NB-2000 requirements.

Table 2.6.5

LIST OF ASME CODE ALTERNATIVES FOR MULTI-PURPOSE CANISTERS (MPCS)

MPC Enclosure Vessel	NB-3100 NF-3100	Provides requirements for determining design loading conditions, such as pressure, temperature, and mechanical loads.	These requirements are subsumed by the HI-STORM FW FSAR, serving as the Design Specification, which establishes the service conditions and load combinations for the storage system.
MPC Enclosure Vessel	NB-4120	NB-4121.2 and NF-4121.2 provide requirements for repetition of tensile or impact tests for material subjected to heat treatment during fabrication or installation.	In-shop operations of short duration that apply heat to a component, such as plasma cutting of plate stock, welding, machining, and coating are not, unless explicitly stated by the Code, defined as heat treatment operations.
MPC Enclosure Vessel	NB-4220	Requires certain forming tolerances to be met for cylindrical, conical, or spherical shells of a vessel.	The cylindricity measurements on the rolled shells are not specifically recorded in the shop travelers, as would be the case for a Code-stamped pressure vessel. Rather, the requirements on inter-component clearances (such as the MPC-to-transfer cask) are guaranteed through fixture-controlled manufacturing. The fabrication specification and shop procedures ensure that all dimensional design objectives, including inter-component annular clearances are satisfied. The dimensions required to be met in fabrication are chosen to meet the functional requirements of the dry storage components. Thus, although the post-forming Code cylindricity requirements are not evaluated for compliance directly, they are indirectly satisfied (actually exceeded) in the final manufactured components.

Table 2.6.5

LIST OF ASME CODE ALTERNATIVES FOR MULTI-PURPOSE CANISTERS (MPCS)

MPC Enclosure Vessel	NB-4122	Implies that with the exception of studs, bolts, nuts and heat exchanger tubes, CMTRs must be traceable to a specific piece of material in a component.	MPCs are built in lots. Material traceability on raw materials to a heat number and corresponding CMTR is maintained by Holtec through markings on the raw material. Where material is cut or processed, markings are transferred accordingly to assure traceability. As materials are assembled into the lot of MPCs being manufactured, documentation is maintained to identify the heat numbers of materials being used for that item in the multiple MPCs being manufactured under that lot. A specific item within a specific MPC will have a number of heat numbers identified as possibly being used for the item in that particular MPC of which one or more of those heat numbers (and corresponding CMTRS) will have actually been used. All of the heat numbers identified will comply with the requirements for the particular item.
MPC Lid and Closure Ring Welds	NB-4243	Full penetration welds required for Category C Joints (flat head to main shell per NB-3352.3)	MPC lid and closure ring are not full penetration welds. They are welded independently to provide a redundant seal.
MPC Closure Ring, Vent and Drain Cover Plate Welds	NB-5230	Radiographic (RT) or ultrasonic (UT) examination required.	Root (if more than one weld pass is required) and final liquid penetrant examination to be performed in accordance with NB-5245. The closure ring provides independent redundant closure for vent and drain cover plates. Vent and drain port cover plate welds are helium leakage tested.
MPC Lid to	NB-5230	Radiographic (RT) or	Only progressive liquid penetrant

Table 2.6.5

LIST OF ASME CODE ALTERNATIVES FOR MULTI-PURPOSE CANISTERS (MPCS)

Shell Weld		ultrasonic (UT) examination required.	(PT) examination is permitted. PT examination will include the root and final weld layers and each approx. 3/8" of weld depth.
MPC Enclosure Vessel and Lid	NB-6111	All completed pressure retaining systems shall be pressure tested.	The MPC vessel is seal welded in the field following fuel assembly loading. The MPC vessel shall then be pressure tested as defined in Chapter 10. Accessibility for leakage inspections preclude a Code compliant pressure test. All MPC enclosure vessel welds (except closure ring and vent/drain cover plate) are inspected by volumetric examination. MPC shell and shell to baseplate welds are subject to a fabrication helium leak test prior to loading. The MPC lid-to-shell weld shall be verified by progressive PT examination. PT must include the root and final layers and each approximately 3/8 inch of weld depth. The inspection results, including relevant findings (indications) shall be made a permanent part of the user's records by video, photographic, of other means which provide an equivalent record of weld integrity. The video or photographic records should be taken during the final interpretation period described in ASME Section V, Article 6, T-676. The vent/drain cover plate and the closure ring welds are confirmed by liquid penetrant examination. The inspection of the weld must be performed by qualified personnel and shall meet the acceptance

Table 2.6.5

LIST OF ASME CODE ALTERNATIVES FOR MULTI-PURPOSE CANISTERS (MPCS)

			requirements of ASME Code Section III, NB-5350.
MPC Enclosure Vessel	NB-7000	Vessels are required to have overpressure protection.	No overpressure protection is provided. Function of MPC enclosure vessel is to contain radioactive contents under normal, off-normal, and accident conditions of storage. MPC vessel is designed to withstand maximum internal pressure considering 100% fuel rod failure and maximum accident temperatures.
MPC Enclosure Vessel	NB-8000	States requirements for nameplates, stamping and reports per NCA-8000.	The HI-STORM FW System is to be marked and identified in accordance with 10CFR71 and 10CFR72 requirements. Code stamping is not required. QA data package to be in accordance with Holtec approved QA program.

2.7 SAFETY PROTECTION SYSTEMS

2.7.1 General

The HI-STORM UMAX System is engineered to provide for the safe long-term storage of spent nuclear fuel (SNF). The HI-STORM UMAX will withstand all normal, off-normal, and postulated accident conditions without release of radioactive material or excessive radiation exposure to workers or members of the public. Special considerations in the design have been made to ensure long-term integrity and confinement of the stored SNF throughout all cask normal and off-normal operating conditions and its retrievability for further processing or ultimate disposal in accordance with 10 CFR 72.122(l) and ISG-2 [2.7.1].

The HI-STORM UMAX System is completely passive requiring no active components or instrumentation to perform its design functions. Temperature monitoring or scheduled visual verification of the integrity of the air passages is used to verify continued operability of the VVM heat removal system, as set down in the system's Technical Specification.

2.7.2 Protection by Multiple Confinement Barriers and Systems

a. Confinement Barriers and Systems

The confinement of the spent fuel is provided by the MPC's Enclosure Vessel. .

Contamination on the outside of the MPC from the fuel pool water is minimized by preventing contact, removing the contaminated water, and decontamination. An inflatable seal in the annular gap between the MPC and HI-TRAC, and the elastomer seal in the HI-TRAC bottom lid (see Chapter 9 of the HI-STORM FW FSAR) prevent the fuel pool water from contacting the exterior of the MPC and interior of the HI-TRAC VW while submerged for fuel loading.

The MPC is a seal welded enclosure which provides the confinement boundary. The MPC confinement boundary is defined by the MPC baseplate, MPC shell, MPC lid, closure ring, port cover plates, and associated welds.

The MPC confinement boundary has been designed to withstand any postulated off-normal operations, accident conditions, or external natural phenomena. Redundant closure of the MPC is provided by the MPC closure ring welds which provide a second barrier to the release of radioactive material from the MPC internal cavity. Therefore, no monitoring system for the confinement boundary is required.

Confinement is discussed further in Chapter 7 of FW FSAR. MPC field weld examinations, helium leakage testing of the port cover plates welds, and pressure testing are performed to verify the confinement function. Fabrication inspections and tests are also performed, as discussed in Chapter 10 of the HI-STORM FW FSAR, to verify the integrity of the confinement boundary.

b. Cask Cooling

To ensure that an effective passive heat removal capability exists for long-term satisfactory performance, several thermal design features are incorporated in the storage system. They are as follows:

The MPC fuel basket is formed by a honeycomb structure of Metamic-HT plates which allows the unimpeded conduction of heat from the center of the basket to the periphery. The MPC cavity is equipped with the capability to circulate helium internally by natural buoyancy effects and transport heat from the interior region of the canister to the peripheral region (Holtec Patent 5,898,747).

The MPC confinement boundary ensures that the inert gas (helium) atmosphere inside the MPC is maintained during normal, off-normal, and accident conditions of storage and transfer. The MPC confinement boundary maintains the helium confinement atmosphere below the design temperatures and pressures stated in Table 2.3.7 and Table 2.3.5, respectively.

The MPC thermal design maintains the fuel rod cladding temperatures below the ISG-11 limits such that fuel cladding does not experience degradation during the long term storage period.

The HI-STORM UMAX is optimally designed, with multiple cooling passages and suitably sized flow annuli, which maximize air flow by ensuring a turbulent flow regime at Design Basis heat loads.

As shown in the licensing drawing package, cooling air to each MPC storage cavity is provided by four independent ducts. Thus, there is a significant level of redundancy in the cooling air delivery system for the HI-STORM UMAX.

As can be observed from the licensing drawings, the air inlet locations are separated from the outlet vent by a significant lateral and vertical distance. This design feature ensures that there is minimal mixing of cold and heated air in the storage system. Calculations summarized in Chapter 4 show that the heat rejection performance of the system is stable under varying wind speed.

2.7.3 Protection by Equipment and Instrumentation Selection

a. Equipment

The HI-STORM UMAX System may include use of ancillary or support equipment for ISFSI implementation. Ancillary equipment and structures utilized at the HI-STORM UMAX ISFSI may be broken down into two broad categories, namely Important-to-Safety (ITS) ancillary equipment and Not Important to Safety (NITS) ancillary equipment. NUREG/CR-6407 provides guidance for the determination of a component's safety classification [2.6.4].

The only ancillary equipment used in conjunction with the MPC loading at an ISFSI consists of the Mating Device (a patented design, see Table 1.3.2) and the load handling device such as the cask transporter.

The MPC transfer is carried out by actuating the Mating Device and moving the MPC vertically to the cylindrical cavity of the recipient VVM cavity. The mating device is actuated by removing the bottom lid of the HI-TRAC transfer cask. The device utilized to lift the HI-TRAC transfer cask to place it on the VVM and to vertically transfer the MPC may be of stationary or mobile

type, but it must have redundant drop protection features. The cask transporter can serve as the load handling device.

b. Instrumentation

As a consequence of the passive nature of the HI-STORM UMAX System, Important-to-Safety instrumentation is not necessary. No instrumentation is required or provided for HI-STORM UMAX storage operations, other than normal security service instruments and dosimeters.

However, in lieu of performing the periodic inspection of the HI-STORM UMAX VVM vent screens, temperature elements may be installed inside the VVM outlet duct and below the bottom of outlet screen to continuously monitor the air temperature. If the temperature elements and associated temperature monitoring instrumentation are used as the sole means of surveillance then they shall be designated as Important-to-Safety.

2.8NUCLEAR CRITICALITY SAFETY

The criticality safety criteria stipulates that the effective neutron multiplication factor, k_{eff}, including statistical uncertainties and biases, is less than 0.95 for all postulated arrangements of fuel within the cask under all credible conditions.

2.8.1 Control Methods for Prevention of Criticality

The control methods and design features used to prevent criticality for all MPC configurations are the following:

- Fuel basket constructed of neutron absorbing material with no potential of detachment, delamination or degradation in long term inert environment of the MPC.
- Favorable geometry provided by the MPC fuel basket.
- A high B-10 concentration (50% greater than the concentration used in the existing state-of-the art designs certified under 10CFR72) leads to a lower reactivity level under all operating scenarios.

Administrative controls shall be used to ensure that fuel placed in the HI-STORM UMAX System meets the requirements described in Chapters 2. All appropriate criticality analyses are presented in Chapter 6 of the HI-STORM FW FSAR.

2.8.2 Error Contingency Criteria

Provision for error contingency is built into the criticality analyses performed in Chapter 6. Because biases and uncertainties are explicitly evaluated in the analysis, it is not necessary to introduce additional contingency for error.

2.8.3 Verification Analyses

In Chapter 6 of the HI-STORM FW FSAR, critical experiments are selected which reflect the design configurations. These critical experiments are evaluated using the same calculation methods, and a suitable bias is incorporated in the reactivity calculation.

2.9 RADIOLOGICAL PROTECTION

a. Access Control

As required by 10CFR72, uncontrolled access to the ISFSI is prevented through physical protection means. A security fence surrounded by a physical barrier fence with an appropriate locking and monitoring system is a standard approach to limit access if the ISFSI is located outside the controlled area. The details of the access control systems and procedures, including division of the site into radiation protection areas, will be developed by the licensee (user) of the ISFSI utilizing the HI-STORM UMAX System.

b. Shielding

The objective of shielding is to assure that radiation dose rates at key locations are as low as practical in order to maintain occupational doses to operating personnel As Low As Reasonably Achievable (ALARA) and to meet the requirements of 10 CFR 72.104 and 10 CFR 72.106 for dose at the controlled area boundary (see Table 2.9.1).

The HI-STORM UMAX is designed to limit dose rates in accordance with 10CFR72.104 and 10CFR72.106 which provide radiation dose limits for any real individual located at or beyond the nearest boundary of the controlled area. The individual must not receive doses in excess of the limits given in Table 2.9.1 for normal, off-normal, and accident conditions.

Three locations are of particular interest in the storage mode:

- immediate vicinity of the cask
- restricted area boundary
- controlled area (site) boundary

Dose rates in the immediate vicinity of the loaded VVM are important in consideration of occupational exposure. Conservative evaluations of dose rate have been performed and are described in Chapter 5 based on Reference PWR fuel.

Consistent with 10 CFR 72, there is no single dose rate limit established for the HI-STORM UMAX System. Compliance with the regulatory limits on occupational and controlled area doses is performance-based, as demonstrated by dose monitoring performed by each cask user.

Design objective dose rates for the HI-STORM UMAX system are presented in Table 2.9.2.

Because of the passive nature of the HI-STORM UMAX System, human activity related to the system after deployment in storage is infrequent and of short duration. Personnel exposures due to operational and maintenance activities are discussed in Chapter 11, wherein measures to reduce occupational dose are also discussed. The estimated occupational doses for personnel provided in Chapter 11 comply with the requirements of 10CFR20. As discussed in Chapter 11, the HI-STORM UMAX System has been configured to minimize both the site boundary dose in storage and occupational dose during short-term operations to the maximum extent possible.

The analyses and discussions presented in Chapters 5, 9, and 11 demonstrate that the HI-STORM UMAX System is capable of meeting the radiation dose limits set down in Table 2.9.1.

c. Radiological Alarm System

The HI-STORM UMAX does not require a radiological alarm system. There are no credible events that could result in release of radioactive materials from the system. Furthermore, direct radiation exposure from the ISFSI is subject to monitoring by the plant's existing dose monitoring system.

Table 2.9.1 RADIOLOGICAL SITE BOUNDARY REQUIREMENTS	
MINIMUM DISTANCE TO BOUNDARY OF CONTROLLED AREA (m)	100
NORMAL AND OFF-NORMAL CONDITIONS:	
-Whole Body (mrem/yr)	25
-Thyroid (mrem/yr)	75
-Any Other Critical Organ (mrem/yr)	25
DESIGN BASIS ACCIDENT:	
-TEDE (rem)	5
-DDE + CDE to any individual organ or tissue (other than lens of the eye) (rem)	50
-Lens dose equivalent (rem)	15
-Shallow dose equivalent to skin or any extremity (rem)	50

Table 2.9.2

DESIGN OBJECTIVE DOSE RATES FOR HI-STORM UMAX

Area of Interest	Dose Rate (mrem/hr)
Inlet Vents	80
Inlet Plenum Region	150
Outlet Vent	60

2.10 FIRE AND EXPLOSION PROTECTION

There are no combustible or explosive materials associated with the HI-STORM UMAX System. Combustible materials will not be stored within the HI-STORM UMAX ISFSI. However, for conservatism, a hypothetical fire accident has been analyzed in Chapter 4 as a bounding condition for HI-STORM UMAX System. The evaluation of the HI-STORM UMAX System fire accident is discussed in Chapter 12.

Explosive material will not be stored within an ISFSI. Small overpressures may result from accidents involving explosive materials which are stored or transported in the vicinity of the site. Explosion as an accident loading condition has been considered in Chapter 12.

2.11 DECOMMISSIONING CONSIDERATIONS

Efficient decommissioning of the ISFSI is a paramount objective of the HI-STORM UMAX System. The HI-STORM UMAX System is ideally configured to facilitate rapid, safe, and economical decommissioning of the storage site. As discussed below, Holtec International has taken appropriate steps to ensure that the necessary equipment designs and certifications shall be available to the user of the HI-STORM UMAX System to expeditiously decommission the ISFSI at the end of the storage facility's required service life.

Towards that end, the loaded MPC has been designed with the objective to transport it in a HI-STAR 190 transportation cask (Figure 2.11.1) Since the loaded MPC is a self-contained Waste Package, no further handling of the SNF stored in the MPC will be required prior to transport to a licensed centralized storage facility or repository.

The MPC which holds the SNF assemblies is engineered to be suitable as a waste package for permanent internment in a deep Mined Geological Disposal System (MGDS). The materials of construction permitted for the MPC are known to be highly resistant to severe environmental conditions. No carbon steel, paint, or coatings are used or permitted in the MPC in areas where they could be exposed to spent fuel pool water or the ambient environment. Therefore, the SNF assemblies stored in the MPC do not need to be removed. However, to ensure a practical, feasible method to defuel the MPC, the top of the MPC is equipped with sufficient gamma shielding and markings locating the drain and vent locations to enable semiautomatic (or remotely actuated) severing of the MPC closure ring to provide access to the MPC vent and drain. The circumferential welds of the MPC closure lid can be removed by semiautomatic or remotely actuated means, providing access to the SNF.

Likewise, the VVM consists of steel and concrete rendering it suitable for permanent burial. Alternatively, the MPC can be removed from the VVM, and the latter reused for storage of other MPCs. In either case, the VVM would be expected to have no interior or exterior radioactive surface contamination. Any neutron activation of the steel and concrete is expected to be extremely small, and the assembly would qualify as Class A waste in a stable form based on definitions and requirements in 10CFR61.55. As such, the material would be suitable for burial in a near-surface disposal site as Low Specific Activity (LSA) material.

If the SNF needs to be removed from the MPC before it is placed into the MGDS, the MPC interior metal surfaces can be decontaminated using existing mechanical or chemical methods to allow for its disposal. This will be facilitated by the smooth metal surfaces designed to minimize crud traps. After the surface contamination is removed, the MPC radioactivity will be diminished significantly, allowing near-surface burial or secondary applications at the licensee's facility.

The HI-STORM UMAX ISFSI will contain MPCs that are readily removable from their storage cavities. The ISFSI will be decommissioned only after all MPCs have been removed from the storage cavities. Removing the Divider Shell does not require any weld removal or unfastening

of bolts. The CEC structure can be removed by excavating the surrounding subgrade. Alternatively, the cavity can be filled with suitable fill materials and the CEC left in place. Even if the decision is made to dispose of all activated material, the VVM, due to differences in its geometry and construction (particularly, use of the native soil as the biological shield to the extent possible) will result in less steel and concrete to be disposed off. In the aggregate, it is estimated that less material will need to be disposed off to decommission a VVM ISFSI in comparison to an ISFSI containing aboveground VVMs.

Finally, the activation estimate in Table 2.4.1 of [2.6.3] for the aboveground VVM inner shell is adopted herein (conservatively) for the VVM steel shell enclosure.

Due to the design of the HI-STORM UMAX System, no residual contamination is expected to be left behind on the concrete ISFSI pad. The base pad, fence, and peripheral utility structures will require no decontamination or special handling after the last VVM is removed.

The long-lived radio-nuclides produced by the irradiation of the HI-STORM UMAX System components are listed in Table 2.11.1. The activation of the HI-STORM UMAX components shall be limited to a cumulative activity of 10 Ci per cubic meter before decommissioning and disposal of the activated item can be carried out.

Table 2.11.1

PRINCIPAL LONG-LIVED ISOTOPES PRODUCED DURING IRRADIATION OF THE HI-STORM UMAX COMPONENTS

Nuclide	MPC Stainless Steel	HI-STORM Steel	HI-STORM Concrete
^{54}Mn	X	X	X
^{55}Fe	X	X	X
^{59}Ni	X	-	-
^{60}Co	X	-	-
^{63}Ni	X	-	-
^{39}Ar	-	-	X
^{41}Ca	-	-	X

[PROPRIETARY INFORMATION WITHHELD IN ACCORDANCE WITH 10 CFR 2.390]

Figure 2.11.1: HI-STAR 190 Transportation Overpack and MPC Shown in Exploded, Cut-Away View

2.12 REGULATORY COMPLIANCE

Pursuant to the guidance provided in NUREG-1536, the foregoing material in this Chapter provides:

A complete set of principal design criteria for the VVM as mandated by 10CFR72.24I(1), §72.24(c)(2), §72.120(a) and §72.236(b);

A clear identification of VVM structural parts subject to a fully articulated design subject to certification under 10CFR72 and of interfacing structures;

The required set of limiting critical characteristics of interfacing ISFSI Structures to ensure that the VVM will render its intended function under all design basis scenarios of operation; and

A complete set of requirements premised on well-recognized codes and standards to govern the design and analysis (to establish safety margins) and manufacturing of the VVM.

It is noted that the requirements of 10CFR72 do not preclude the use of an underground storage system such as the HI-STORM UMAX. The underground VVM design, while not specifically mentioned in the regulatory guidance literature associated with implementing the requirements in 10CFR72 (i.e., NUREG-1536) meets and exceeds the intent of the guidance in that it provides an enhanced protection of the stored spent nuclear fuel and a significantly reduced site boundary dose, enables a more convenient handling operation, and presents a much smaller target for missiles/projectiles compared to an aboveground storage system.

2.13 REFERENCES

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CHAPTER 3: STRUCTURAL EVALUATION[1]

3.0 OVERVIEW

In this chapter, the structural safety evaluation of the HI-STORM UMAX Vertical Ventilated Module (VVM) system is performed pursuant to the guidelines of NUREG-1536. The organization of technical information in this chapter mirrors the format and content of Regulatory Guide 3.61 to the extent applicable. The objective of the structural analyses is to ensure that the integrity of the HI-STORM UMAX system is maintained under the normal, off-normal and extreme environmental conditions and accident events listed in Chapter 2. The design basis information contained in the previous two chapters and in this chapter provides the necessary data to permit all needed structural evaluations for demonstrating compliance with the requirements of 10CFR72.24. To facilitate regulatory review, the assumptions and conservatisms inherent in the analyses are identified along with a concise description of the analytical methods, models, and acceptance criteria. A summary of the system's ability to maintain its structural integrity under other effects that may contribute to structural failure, such as fatigue, buckling, and non-ductile fracture is also provided. (An evaluation of the suitability of the materials used in the HI-STORM UMAX VVM system under both slow acting (degenerative) and fast acting (precipitous) loads is presented in Chapter 8.)

The VVM, consisting of the CEC and the Closure Lid, serves as the storage space for the loaded MPC. The CEC is a weldment of the Container Shell, Container Flange, Bottom Plate, Lower MPC Guides, and MPC Bearing Pads. The Closure Lid is a weldment of structural steel encasing plain concrete and arranged to provide an appropriate outlet passage for the heated air issuing from the storage cavity. An insulated Divider Shell with Upper MPC Guides is situated within the CEC and restrained by the Lower MPC Guides at the bottom and by the Container Flange at the top. These individual components are collectively referred to as VVM Components. Interfacing structures that surround and support the VVM, as well as proximate structures, which are collectively referred to as ISFSI Structures, are explained in Chapter 2. Section 1.2 contains a complete description of the VVM components and ISFSI Structures (accompanied by appropriate figures) and their respective functions within the HI-STORM UMAX ISFSI. The essential design details of both the VVM Components and the ISFSI Structures are set down in the Licensing Drawing in Section 1.5. The design basis loadings for the facility are provided in Chapter 2. The applicable codes, standards, and practices governing the structural analysis of the HI-STORM UMAX module, as well as the design criteria, are also presented in Chapter 2. Throughout this chapter, in the context of the VVM components, the term "*safety factor*" is defined as the *ratio of the allowable stress (load) or displacement for the applicable load combination to the maximum computed stress (load) or displacement.*

For the ISFSI Structures, which are made of reinforced concrete, the safety factor is defined as the ratio of the ultimate moment (or shear) capacity to the actual maximum moment (or shear)

1 The analyses summarized in this chapter focus on the governing canisters out of the population of MPCs listed in Table 1.2.1. Specifically, this chapter supports the certification of only MPC-37 and MPC-89 at this time. The analyses reported for smaller canisters are for reference purposes only.

developed under the factored load combination.

The following information germane to safety factors is important to understanding the safety case presented in this chapter:

1. All safety factors are dimensionless.

2. The minimum permissible value of any safety factor to support a positive safety conclusion is 1.0. However, to permit an occasional 10CFR72.48 evaluation that may be required, the HI-STORM UMAX system is designed to ensure that all safety factors in the initial design basis calculations are at least 1.1.

The objective of the structural analyses is to ensure that the integrity of the HI-STORM UMAX system is maintained under all credible loadings under normal, off-normal and extreme environmental conditions as well all credible accident events. Specifically, the design basis information contained in the previous two chapters and in this chapter provides the necessary data to permit all needed structural evaluations for demonstrating compliance with the requirements of 10CFR72.236(a), (b), (d) (e), (f), (g), and (1). To facilitate regulatory review, the assumptions and conservatisms inherent in the analyses are identified along with a concise description of the analytical methods, models, and acceptance criteria. The information presented herein is intended to comply with the guidelines of NUREG-1536 and ISG-21 pertaining to use of finite element codes.

In particular, every Computational Modeling Software (CMS) deployed to perform the structural analyses is identified and its implementation appropriately justified as suggested in ISG-21. The information on benchmarking and validation of each Computational Modeling Software is also provided (in Subsection 3.6.2).

Where appropriate, the structural analyses have been performed using classical strength materials solution. Such calculations are presented in this FSAR in transparent detail.

Finally, the input data and analyses using Computational Modeling Software (CMS) are described in sufficient detail to enable an independent evaluation of safety conclusions reached in this chapter.

The MPC and HI-TRAC are categorized as "interfacing components" in this FSAR because they have been licensed in their host docket (i.e., docket number 72-1032 for HI-STORM FW). Nevertheless, it is necessary to demonstrate in this FSAR that the loads exerted on an interfacing component do not exceed its licensing basis in its docket number 72-1032. For example, the MPC structural integrity has been evaluated in this chapter to ensure that the rattling motion of the MPC inside the VVM storage cavity during the DBE event will not produce a breach in the confinement boundary or subject the MPC to lateral decelerations in excess of the value for which it was qualified in docket number 72-1032.

However, it should be noted without ambiguity that the HI-STORM UMAX system has been designed such that, with the exception of the earthquake induced impact load on its Confinement Boundary,

the MPC stored in HI-STORM UMAX will always be subject to a less severe loading than what would be obtained in the above-ground HI-STORM systems under an identical environmental condition. Furthermore, the following restrictions apply:

1. *The content conditions for an MPC to be stored in HI-STORM UMAX must not exceed the technical specification for HI-STORM UMAX.*
2. *The design specifications for the MPC (and the HI-TRAC cask used to load the MPC that are identified in its licensing basis FSAR) will apply at all times during loading and storage of the MPC. Such specifications include Design Basis external pressure and any acceptable radiation dose limits.*

Table 3.1.1 provides the acceptance criteria that apply to the MPCs listed in Table 1.2.1.

Technical descriptions and safety analyses in this chapter, especially pertaining to the components common to the HI-STORM FW and the HI-STORM UMAX systems are referenced in this FSAR, as necessary to the HI-STORM FW FSAR. To facilitate convenient access to the referenced material, the latest edition of the HI-STORM FW FSAR has been placed in this docket and a list of HI-STORM FW FSAR sections germane to this chapter is provided in a tabular form below.

The safety analyses summarized in this chapter demonstrate acceptable margins to the allowable limits under all design basis loading conditions and operational modes. Minor changes to the design parameters that inevitably occur during the product's life cycle which are treated within the purview of 10CFR72.48 and are ascertained to have an insignificant effect on the computed safety factors may not prompt a formal reanalysis and revision of the results and associated data in the tables of this chapter unless the cumulative effect of all such unquantified changes on the reduction of any of the computed safety margins cannot be deemed to be insignificant. For purposes of this determination, an insignificant loss of safety margin with reference to an acceptance criterion is defined as the estimated reduction that is no more than one order of magnitude below the available margin reported in the FSAR. To ensure rigorous configuration control, the information in the Licensing drawings in Section 1.5 should be treated as the authoritative source for numerical analysis at all times. Reliance on the input data and associated results in this chapter for additional mathematical computations may not be appropriate as they serve the sole purpose of establishing safety compliance in accordance with the acceptance criteria set down in Chapter 2 and in this chapter.

HI-STORM FW FSAR MATERIAL ADOPTED IN THIS FSAR BY REFERENCE [2]		
Location of UMAX FSAR	**Subject of the reference**	**Location in HI-STORM FW FSAR, Revision 3**
Subsection 3.1.3	Structural Qualifications of MPC	Subsections 3.4.3 and 3.4.4
Subsection 3.1.3	Structural Qualifications of HI-TRAC	Subsections 3.4.3 and 3.4.4
Subsection 3.4.4	MPC Qualified for accident conditions of storage	Subsection 3.4.4
Subsection 3.4.4	Stress Analysis for MPCs and HI-TRAC under normal handling conditions	Subsection 3.4.3

2 For convenience of reference, the specific revision of the HI-STORM FW FSAR that is referenced in the safety analysis herein is placed in this docket. Updated versions of the HI-STORM FW FSAR shall be placed in this docket as necessary so as to ensure that the safety analyses on the “UMAX” docket (72-1040) remain aligned with the material referenced in the HI-STORM FW FSAR.

3.1 STRUCTURAL DESIGN

3.1.1 Overview

This chapter presents the structural evaluation of the VVM Components for the applicable load cases summarized in Chapter 2 (Table 2.4.1). In Section 3.4, the safety factors for each load case for the VVM Components are quantified. In addition, the safety evaluation of the ISFSI Structures is carried out using the factored load combinations from ACI-318(2005) [2.6.2] (see Table 2.4.3). Summary tables of bounding safety factors are provided for governing load combination for the ISFSI Structures. The Licensing Drawing for the HI-STORM UMAX VVM is provided in Section 1.5. Section 2.6 provides a summary of the applicable regulations and codes and standards for the VVM Components and the ISFSI structures. The design of the VVM Components and the ISFSI Structures is depicted in sufficient detail in the Licensing Drawing to enable the safety analyses summarized in this chapter to be performed. The applicable Design Basis Earthquake is defined by the response spectra shown in Figures 2.4.1 and 2.4.2.

3.1.2 Design Criteria and Applicable Loads

Consistent with the provisions of NUREG-1536, the central objective of the structural analysis presented in this chapter is to ensure that the HI-STORM UMAX system possesses sufficient structural capacity to withstand normal and off-normal loads and the worst case loads under extreme environmental phenomenon or accident events. Withstanding such loadings implies that the HI-STORM UMAX system must successfully preclude the following:

- unacceptable risk of criticality
- unacceptable release of radioactive materials
- unacceptable radiation levels
- impairment of ready retrievability of the SNF

The above design objectives for the HI-STORM UMAX system can be particularized for individual components as follows:

- The objective of the structural analysis of the MPC is to demonstrate that:

  i. Confinement of radioactive material is maintained under normal, off-normal, accident conditions, and natural phenomenon events.

  ii. The MPC basket does not deform under credible loading conditions such that the subcriticality or retrievability of the SNF is jeopardized.

  As stated previously in Section 3.0, the certification of a new MPC design is not envisaged for this docket. Any MPC that is to be stored in HI-STORM UMAX must have been certified in another docket.

Furthermore, the design of the HI-STORM UMAX VVM has been configured such that it maintains the steady state stresses acting on the MPC at a lower (equal) level than those that would develop at the design basis heat load during storage in the above-ground HI-STORM FW overpack in docket number 72-1032. Among the environmental loadings, the case of earthquakes warrants special attention because the manner of support provided to the MPC in HI-STORM UMAX is quite different from the above ground overpacks. To ensure that the MPC design continues to meet the above safety goals under the earthquake event, acceptance criteria for the MPC in Table 3.1.1 have been provided.

- The objectives of the structural analysis of the storage modules are to demonstrate that:

  i. Large energetic missiles such as tornado-generated missiles (see Subsection 2.4.2) do not compromise the integrity of the MPC Confinement Boundary.

  ii. The radiation shielding remains properly positioned in the case of any normal, off-normal, or natural phenomenon or accident event.

  iii. The flow path for the cooling airflow shall remain available under normal and off-normal conditions of storage and after an extreme environmental phenomenon or an accident event.

  iv. The loads arising from normal, off-normal, and accident level conditions exerted on the contained MPC do not violate the structural design criteria of the MPC.

  v. No geometry changes occur under any normal, off-normal, and accident level conditions of storage that preclude ready retrievability of the contained MPC.

  vi. The inter-cask transfer of a loaded MPC can be carried out without exceeding the structural capacity of the HI-STORM UMAX module.

Design (and acceptance) criteria for the HI-STORM UMAX VVM components and the ISFSI structures are summarized in Tables 2.3.1 and 2.3.2. The acceptance criteria for the MPC are provided in Table 3.1.1.

3.1.2.1 Applicable Loadings

Individual loads, applicable to the HI-STORM UMAX System, are defined in Sections 2.4 and 2.5.

3.1.2.2 Design Basis Loads and Load Combinations

i. Design Basis Loads and Load Combinations for VVM Components

Load combinations are developed by assembling the individual loads that may act concurrently, and possibly, synergistically. Load cases and design basis loads applicable to the VVM Components are summarized in Table 2.4.1. Those load cases in Table 2.4.1 which involve an "interfacing component" (namely the stored MPC and the HI-TRAC transfer cask) are analyzed with an

appropriate model of the interfacing component included in the simulation of the storage system. For example, the evaluation of the design basis earthquake (DBE) loading includes the MPC. Results of the analyses carried out under the Design Basis Loads are compared with their respective allowable limits and/or functional performance criteria, as applicable.

ii. Design Basis Loads and Load Combinations for ISFSI Structures

The HI-STORM UMAX ISFSI consists of plate-type reinforced concrete structures whose minimum section strength properties are defined by Table 2.3.2 and the Licensing Drawings. The ISFSI is supported by the subgrade underneath the SFP, which may include pilings, if required, to meet the effective shear wave velocity in Table 2.3.2. Table 2.4.3 contains load combinations applicable to the ISFSI Structures (reinforced concrete structures) in the HI-STORM UMAX ISFSI. The individual loadings on the ISFSI are:

a. Dead load of the VVM and the concomitant effect of settlement over the Design Life of the system (D in Table 2.4.3). The method to incorporate the effect of long-term settlement of the subgrade underneath the SFP (also referred to as the “undergrade”), described in Section 3.4, is used. This method essentially consists of using the deflection properties of the different layers to define equivalent elastic properties of the subgrade underneath the SFP. In the finite element analysis of the SFP, the equivalent (degraded) elastic properties of the subgrade underneath the SFP are utilized to account for the effect of long-term settlement. The Dead load on the SFP from the weight of the loaded VVMs nearly equals the weight of the earth removed. Therefore, the long-term settlement of the SFP is expected to be quite small. The ISFSI pad will not settle relative to the SFP over its service life since it’s supported by the Self-hardening Engineered Subgrade (SES).

b. Live load from the loaded transporter acts directly on the ISFSI pad. This load also adds to the overall load on the SFP (L in Table 2.4.3). The load from the loaded transporter is the sole live load applicable to the ISFSI structures.

c. Seismic load is computed using the methodology presented in Subsection 3.4.4. This load, denoted as E in Table 2.4.3, is the aggregate of the peak dynamic load exerted on the ISFSI less the dead weight. For conservatism, the load E is applied as a static load in the stress analysis of ISFSI structures even though it is transient in nature.

Finally, the subgrade (the space between the ISFSI pad and the Support Foundation Pad) is designated as ITS Category C (Table 2.6.2) to insure that it renders its intended function under normal as well as extreme environmental conditions. Accordingly, numerical values for the required properties such as (spatial average) minimum density, shear wave velocity (Table 2.3.2) and key structural strength values (Table 3.3.4) are provided to enable a comprehensive stress analysis. The switch to CLSM or lean concrete for the "UMAX" ISFSI is, in contrast to HI-STORM 100U which permitted native soil, motivated by considerations to make the ISFSI a competent structure under design basis earthquake and other environmental loadings. The structural analyses documented in this chapter show that, without crediting an Enclosure Wall (even if present):

1. The subgrade in Space A (Figure 2.4.4) will not collapse or otherwise fail causing streaming of radiation to the environment if the ISFSI were subject to a Design Basis Earthquake or impacted by a Design Basis Missile defined in Chapter 2.
2. The ISFSI will remain stable if subjected to the Design Basis Earthquake and if its external subgrade on any side has been excavated down to the SFP (i.e., no lateral support from the subgrade outside the limits of the ISFSI).
3. Even with the contiguous subgrade excavated down to the support Foundation Pad elevation, a Design Basis Missile impacting the subgrade laterally will fail to access the stored MPC.
4. The MPC will remain retrievable under all design basis loading events.

3.1.2.3 Allowables

The important-to-safety components of the HI-STORM UMAX system are identified on the drawings in Section 1.5. Allowable stresses, as appropriate, are tabulated for these components for all service conditions.

Relationships for allowable stresses and stress intensities for NB [3.1.4] and NF [3.1.5] components are provided in Tables 3.1.11 and 3.1.12, respectively. Tables 3.1.2 through 3.1.8 contain numerical values of the stresses/stress intensities for all MPC and VVM load bearing Code materials as a function of temperature.

In all tables the terms S, S_m, S_y, and S_u, respectively, denote the design stress, design stress intensity, minimum yield strength, and the ultimate strength. Property values at intermediate temperatures that are not reported in the ASME Code are obtained by linear interpolation. Property values are not extrapolated beyond the limits of the Code in any structural calculation.

Additional terms relevant to the stress analysis of the HI-STORM UMAX system extracted from the ASME Code (see Figure NB-3222-1, for example) are listed in Table 3.1.10.

3.1.2.4 Brittle Fracture

HI-STORM UMAX utilizes the same material types as HI-STORM 100 (Docket number 72-1014) and HI-STORM FW (docket number 72-1032). In addition, the material thicknesses used in HI-STORM UMAX do not exceed those in the previously certified HI-STORM modules. Furthermore, in comparison to its above ground counterparts, the lowest material temperatures in the HI-STORM UMAX shell, because of its underground configuration, will not be as severely affected by the cold ambient temperatures in winter months. Therefore, the safety considerations for brittle fracture for the HI-STORM UMAX structural steel materials are informed (and bounded) by those previously used in the above-ground HI-STORM modules (see also Section 8.4). Table 3.1.9 reproduces the applicable fracture toughness test requirements for the HI-STORM UMAX VVM from the HI-STORM FW FSAR.

3.1.2.5 Fatigue

Fatigue is a consequence of a cyclic state of stress applied on a metal part. Failure from fatigue occurs if the combination of amplitude of the cyclic stress, σ_a, and the number of cycles, n_f, reaches a threshold value at which failure occurs. ASME Code, Section III, Subsection NCA provides the σ_a-n_f curves for a number of material types. At $n_f = 10^6$, the required σ_a is referred to as the "Endurance Limit". The Endurance Limit for stainless steel (the material used in the MPC) according to the ASME Code, Section III, Div. 1, Appendices, Table I.9.2, is approximately 28 ksi.

The causative factors for fatigue expenditure in a non-active system (i.e., no moving parts) such as the HI-STORM UMAX system may be:

i. rapid temperature changes
ii. significant pressure changes

The HI-STORM UMAX system is exposed to the fluctuating thermal state of the ambient environment. Effect of wind and relative humidity also play a role in affecting the temperature of the cask components. However, the most significant effects are the large thermal inertia of the system and the relatively low heat transfer coefficients that act to smooth out the daily temperature cycles.

As a result, the amplitude of the cyclic stresses, to the extent that they are developed, remains orders of magnitude below the cask material's Endurance Limit.

The second causative factor, namely, pressure pulsation, is limited to the only pressure vessel in the system – the MPC. Pressure produces several types of stresses in the MPC (see Table 3.1.10); all of which are equally effective in causing fatigue expenditure in the metal. However, the amplitude of stress from the pressure cycling (due to the changes in the ambient conditions) is quite small and well below the Endurance Limit of the stainless steel material.

Therefore, failure from fatigue is not a credible concern for the HI-STORM UMAX system components.

3.1.2.6 Buckling

Buckling is caused by compressive stress acting on a slender section. In the HI-STORM UMAX system, the steel weldment in the overpack is not slender; its height-to-diameter ratio being less than 2. There is no source of compressive stress except from the self-weight of the shell and the overpack weight of the HI-TRAC transfer cask in the stacked condition, which produces a modest state of compressive stress. The state of a small compressive stress combined with a low slenderness ratio makes the HI-STORM UMAX VVM safe from the buckling mode of failure.

The only instance of a significant in-plane load on a HI-STORM UMAX part is the impact load exerted on the MPC Guides by the lateral rattling of the MPC under the DBE event. Calculations have been performed and summarized in this FSAR that demonstrate that the MPC Guides will not

buckle.

3.1.2.7 Consideration of Manufacturing and Material Deviations

Departure from the assumed values of material properties in the safety analyses can, in certain cases, adversely affect the computed safety margins. Likewise, deviations in manufacturing that inevitably occur in custom fabrication of capital equipment may detract from the safety factors reported in this chapter. In what follows, the method and measures adopted to ensure that deviations in material properties or in the fabricated hardware will not undermine the structural safety conclusions are summarized.

It is noted that the yield and ultimate strengths of materials used in the manufacturing of the HI-STORM UMAX components will typically be greater than that assumed in the structural analyses because the ASME Code mandates all Code materials meet the minimum certified property values set down in the Code tables. Holtec International requires the material supplier to provide a Certified Mill Test Report in the format specified in the Code to ensure compliance of all physical properties of the supplied material with the specified Code minimums.

The above measures make the probability of an actual material strength property to be falling below the assumed value in the structural analysis in this chapter to be non-credible. On the contrary, Holtec's manufacturing experience suggests that the actual properties are likely to be uniformly and substantially greater than the assumed values.

A similarly conservative approach is used to ensure that the fabrication processes do not degrade the computed safety margins. Towards this end, the fabrication documents (drawings, travelers and shop procedures) implement a number of pro-active measures to prevent all known sources of development of a strength-adverse condition, such as:

i. All welding procedures are qualified to yield better physical properties than the Code minimums. All essential variables that affect weld quality are tightly controlled.

ii. Only those craftsmen who have passed the welding skill criteria implemented in the shop are permitted to weld.

iii. A rigorous weld material quality over-check program is employed to ensure that every weld wire spool meets its respective Code specification.

iv. All structural welds are specified as minimums: In practice, most exceed the specified minimums significantly. All primary structural welds are subject to Q.C. over-check and sign-off.

In the event of a deviation that may depress the computed safety margin, a non-conformance report is prepared by the manufacturer and subject to a safety analysis by Holtec International's corporate engineering using the same methodology as that described in this FSAR. The item is accepted only

if the safety evaluation musters part 72.48 acceptance criteria. A complete documentation of the life cycle of the NCR is archived in the Company's Permanent Filing System and shared with the designated system user.

The above processes and measures have been in place at the Holtec Manufacturing Division to ensure that an unacceptable reduction in the safety factors due to variation in material properties and manufacturing processes does not occur. The Company's nuclear manufacturing experience over the past 25 years corroborates the effectiveness of the above measures.

3.1.3 Stress Analysis Models and Computer Codes

To evaluate the effect of loads on the HI-STORM UMAX system components, finite element models for stress and deformation analysis are developed. The essential attributes of the finite element models for the HI-STORM UMAX VVM and the MPC, developed for Design Basis earthquake induced impact analyses, are presented in this subsection. All finite element models are three-dimensional and are prepared to the level of discretization appropriate to the problem to be solved. The models are developed using ANSYS and LS-DYNA general purpose codes, which are described in Subsection 3.6.2.

Pursuant to ISG-21, the description of the computational model for each component addresses the following areas:

- Description of the model, its key attributes and its conservative aspects
- Types of finite elements used and the rationale for their selection
- Material properties and applicable temperature ranges
- Modeling simplifications and their underlying logic

In subsequent subsections, where the finite element models are deployed to analyze the different load cases, the presentation includes the consideration of:

- Geometric compliance of the simulation with the physics of the problem
- Boundary conditions
- Effect of tolerances on the results
- Convergence (numerical) of the solutions reported in this FSAR

The input files prepared to implement the finite element solutions as well as detailed results are archived in the Calculation Package [3.4.1] within the Company's Configuration Control System. Essential portions of the results for each loading case necessary to draw safety conclusions are extracted from the Calculation Packages and reported in this FSAR. Specifically, the results summarized from the finite element solutions in this chapter are self-contained to enable an independent assessment of the system's safety. Input data is provided in tabular form as suggested in ISG-21. For consistency, the following units are employed to document input data throughout this chapter:

- Time: second

- Mass: pound

Four commercial computer programs, all with a well-established history of usage in the nuclear industry, have been utilized to perform structural and mechanical analyses documented in this submittal. These codes are described in Subsection 3.6.2.

3.1.3.1 HI-STORM UMAX VVM

The physical geometry and materials of construction of the HI-STORM UMAX modules are provided in Chapter 1 including the drawings in Section 1.5. The HI-STORM UMAX VVM finite element model is developed for the seismic Soil Structure Interaction (SSI) analysis of the standard 5×5 VVM array (see Figure 1.0.1), which is fully loaded with the tallest and heaviest MPC allowed to be stored in this underground storage system. The key attributes of the HI-STORM UMAX VVM LS-DYNA finite element model are:

i. Shell elements are used to model the divider shell and CEC except for the CEC base plate and MPC pedestals, which are modeled using thick shell and solid elements, respectively. The VVM lid model is conservatively simplified as a rigid solid body to maximize the contact force at the lid/CEC interface. The bounding MPC stored in the VVM is conservatively represented by a rigid cylinder to yield bounding stresses in the VVM structural members and bounding loads for the ISFSI structural components.

ii. Based on the experience gained in developing the HI-STORM 100U (docket number 72-1014) VVM model, the HI-STORM UMAX VVM model is meshed sufficiently fine to capture the primary stresses developed under the Design Basis Earthquake condition.

iii. Except for the divider shell, CEC baseplate and MPC pedestal, which are conservatively assumed to behave linear elastically to maximize the predicted impact loads under the seismic condition, the VVM steel members are represented by their applicable nonlinear elastic-plastic true stress-strain relationships, which can be found directly from [3.1.2] or derived from engineering stress-strain data. The methodology used for obtaining a true stress-strain curve from a set of engineering stress-strain data (e.g., strength properties from [3.3.2]) is provided in [3.1.6], which utilizes the following power law relation to represent the flow curve of metal in the plastic deformation region:

$$\sigma = K\varepsilon^n$$

where n is the strain-hardening exponent and K is the strength coefficient. Table 3.1.14 provides the values of K and n used to model the behavior of the structural materials in LS-DYNA.

The key input data of the VVM model is listed in Table 3.1.13.

3.1.3.2 Multi-Purpose Canister (MPC)

The structural qualification of the MPCs is documented in the HI-STORM FW FSAR. The finite element model of the MPC is needed, however, for two reasons:

i. To determine the structural consequences of the impact of the MPC's "hard points", namely the Top Lid and the Baseplate locations, with the radial guides welded to the CEC under the DBE.

ii. To ensure that the maximum deceleration sustained by each MPC due to their impact with the Guides is less than the acceptable value (see Table 3.1.1).

The LS-DYNA finite element model of the MPC is essentially identical to that developed for the non-mechanistic tipover analysis for HI-STORM FW (docket number 72-1032). The contents of the MPC, i.e., fuel assemblies, fuel basket and basket shims, are explicitly modeled to account for the interaction between the MPC shell and the MPC contents. Moreover, the MPC model employs a refined element grid in the impact regions of the canister. Additional attributes of the finite element model are:

i. The MPC shell and fuel basket are modeled using LS-DYNA thick shell elements while the MPC lid, baseplate and each fuel assembly are modeled using solid elements. The MPC lid-to-shell weld is also explicitly modeled using solid elements. The finite element discretization of the MPC is sufficiently detailed to accurately articulate the primary membrane and bending stresses as well as the secondary stresses at locations of impact.

ii. The material properties of the MPC components are taken based on the calculated bounding temperatures under normal storage condition. The fuel basket is divided into four regions based on the temperature distribution of the basket in order to use more realistic material properties for the finite element analysis; the same approach was used in the HI-STORM FW tipover analysis.

iii. Except for the fuel assembly model, which is assumed to behave linear elastically, all other MPC structural members are characterized by the true stress-strain relationship of the material to accurately determine the actual plastic deformation in the MPC enclosure vessel for the confinement evaluation of the MPC under impact condition.

The key input data of the MPC enclosure vessel model is listed in Table 3.1.15.

3.1.3.3 HI-TRAC Transfer Cask

The structural qualification of the transfer casks is contained in the HI-STORM FW FSAR. There are no new loads that arise from the operations described in Chapter 9. Therefore, no additional analysis of the transfer cask is performed in this chapter. In the LS-DYNA SSI analysis, the loaded HI-TRAC, along with the transfer cask transporter, is represented as a rigid body in the finite element model for conservatism.

Table 3.1.1

ACCEPTANCE CRITERIA FOR MPCs FOR DEPLOYMENT IN HI-STORM UMAX

	ITEM	Limiting Value	**Source**
1.	Maximum permissible deceleration under seismic or mechanical loading condition	There is no design basis deceleration limit on the MPC. However, the computed value (i.e., 61.75 g's) for the non-mechanistic tip-over event in the HI-STORM FW docket can be conservatively used for the safety evaluation.	Section 3.4 of the HI-STORM FW FSAR
2.	Maximum local plastic strain in the confinement boundary shell	0.1	Reference [3.1.3]

Table 3.1.2

DESIGN AND LEVEL A: ALLOWABLE STRESS

Reference Code:	ASME NF
Material:	A36
Service Conditions:	Design and Normal
Item:	Stress

Temp. (Deg. F)	Classification and Value (ksi)		
	S	Membrane Stress	Membrane plus Bending Stress
-20 to 650	16.6	16.6	24.9
700	15.6	15.6	23.4

Notes:

1. S = Maximum allowable stress values from Table 1A of ASME Code, Section II, Part D.
2. Stress classification per Paragraph NF-3260.

Table 3.1.3

LEVEL B: ALLOWABLE STRESS

Reference Code: ASME NF
Material: A36
Service Conditions: Off-Normal
Item: Stress

Temp. (Deg. F)	**Classification and Value (ksi)**	
	Membrane Stress	**Membrane plus Bending Stress**
-20 to 650	22.1	33.1
700	20.7	31.1

Table 3.1.4A

DESIGN AND LEVEL A: ALLOWABLE STRESS

Code: ASME NF
Material: A516 (A515) Grade 70, A350-LF3 (A350-LF2)
Service Conditions: Design and Normal
Item: Allowable Stress

Temp. (Deg. F)	**Classification and Value (ksi)**		
	S	**Membrane Stress**	**Membrane plus Bending Stress**
-20 to 400	20.0	20.0	30.0
500	19.6	19.6	29.4
600	18.4	18.4	27.6
650	17.8	17.8	26.7
700	17.2	17.2	25.8
800	12.0	12.0	18.0

Notes:

1. S = Maximum allowable stress values from Table 1A of ASME Code, Section II, Part D.
2. Stress classification per Paragraph NF-3260.
3. Maximum allowable stress values are the lowest of all values for the candidate materials (A516 (A515) Grade 70, A350-LF3 (A350-LF2)) at corresponding temperature. Calculations can be performed using the allowable stress values from this table or using the values directly taken from the ASME code [3.1.4] for a specific candidate material.

Table 3.1.4B

DESIGN AND LEVEL A: ALLOWABLE STRESS

Code:	ASME NF
Material:	A240-304, A240-304L
Service Conditions:	Design and Normal
Item:	Allowable Stress

Temp. (Deg. F)	**Classification and Value (ksi)**		
	S	**Membrane Stress**	**Membrane plus Bending Stress**
-20 to 300	16.7	16.7	25.05
400	15.8	15.8	23.7
500	14.7	14.7	22.05
600	14.0	14.0	21.0
650	13.7	13.7	20.55
700	13.5	13.5	20.25

Notes:

1. S = Maximum allowable stress values from Table 1A of ASME Code, Section II, Part D.
2. Stress classification per Paragraph NF-3260.
3. Maximum allowable stress values are the lowest of all values for the candidate materials (A240-304, A240-304L) at corresponding temperature. Calculations can be performed using the allowable stress values from this table or using the values directly taken from the ASME code [3.1.4] for a specific candidate material.

Table 3.1.5A

LEVEL B: ALLOWABLE STRESS

Code: ASME NF
Material: A516 (A515) Grade 70, A350-LF3 (A350-LF2)
Service Conditions: Off-Normal
Item: Allowable Stress

Temp. (Deg. F)	**Classification and Value (ksi)**	
	Membrane Stress	**Membrane plus Bending Stress**
-20 to 400	26.6	39.9
500	26.1	39.1
600	24.5	36.7
650	23.7	35.5
700	22.9	34.3

Notes:

1. Maximum allowable stress values are the lowest of all values for the candidate materials (A516 (A515) Grade 70, A350-LF3 (A350-LF2)) at corresponding temperature. Calculations can be performed using the allowable stress values from this table or using the values directly taken from the ASME code [3.1.4] for a specific candidate material.

Table 3.1.5B

LEVEL B: ALLOWABLE STRESS

Code: ASME NF
Material: A240-304, A240-304L
Service Conditions: Off-Normal
Item: Allowable Stress

Temp. (Deg. F)	Classification and Value (ksi)	
	Membrane Stress	Membrane plus Bending Stress
-20 to 300	22.2	33.3
400	21.0	31.5
500	19.5	29.3
600	18.6	27.9
650	18.2	27.3
700	17.9	26.9

Notes:

2. Maximum allowable stress values are the lowest of all values for the candidate materials (A240-304, A240-304L) at corresponding temperature. Calculations can be performed using the allowable stress values from this table or using the values directly taken from the ASME code [3.1.4] for a specific candidate material.

Table 3.1.6A

LEVEL D: ALLOWABLE STRESS INTENSITY

Code:	ASME NF
Material:	A516 (A515) Grade 70
Service Conditions:	Accident
Item:	Stress Intensity

Temp. (Deg. F)	**Classification and Value (ksi)**		
	$\mathbf{S_m}$	$\mathbf{P_m}$ AMAX ($1.2S_y$, $1.5S_m$), but < 0.7 S_u	$\mathbf{P_m + P_b}$ 150% of P_m
-20 to 100	23.3	45.6	68.4
200	23.2	41.8	62.7
300	22.4	40.3	60.4
400	21.6	39.0	58.5
500	20.6	37.2	55.8
600	19.4	34.9	52.4
650	18.8	33.8	50.7
700	18.1	32.9	49.4

Notes:

1. Level D allowable stress intensities per Appendix F, Paragraph F-1332.
2. S_m = Stress intensity values per Table 2A of ASME, Section II, Part D.
3. P_m and P_b denote Primary Membrane and Primary Bending Stress, respectively.

Table 3.1.6B

LEVEL D: ALLOWABLE STRESS INTENSITY

Code:	ASME NF
Material:	A240-304, A240-304L
Service Conditions:	Accident
Item:	Stress Intensity

Temp. (Deg. F)	**Classification and Value (ksi)**		
	S_m	**P_m** AMAX ($1.2S_y$, $1.5S_m$), but < 0.7 S_u	**$P_m + P_b$** 150% of P_m
-20 to 100	16.7	30.0	45.0
200	16.7	25.6	38.5
300	16.7	25.0	37.5
400	15.8	23.7	35.5
500	14.7	22.0	33.0
600	14.0	21.0	31.5
650	13.7	20.5	30.8
700	13.5	20.2	30.3

Notes:

1. Level D allowable stress intensities per Appendix F, Paragraph F-1332.
2. S_m = Stress intensity values per Table 2A of ASME, Section II, Part D.
3. P_m and P_b denote Primary Membrane and Primary Bending Stress, respectively.

Table 3.1.7

DESIGN, LEVELS A AND B: ALLOWABLE STRESS INTENSITY

Code:	ASME NB
Material:	Alloy X
Service Conditions:	Design, Levels A and B (Normal and Off-Normal)
Item:	Stress Intensity

Temp. (Deg. F)	Classification and Numerical Value					
	S_m	P_m †	P_L †	$P_L + P_b$ †	$P_L + P_b + Q$ ††	P_e ††
-20 to 100	20.0	20.0	30.0	30.0	60.0	60.0
200	20.0	20.0	30.0	30.0	60.0	60.0
300	20.0	20.0	30.0	30.0	60.0	60.0
400	18.6	18.6	27.9	27.9	55.8	55.8
500	17.5	17.5	26.3	26.3	52.5	52.5
600	16.5	16.5	24.75	24.75	49.5	49.5
650	16.0	16.0	24.0	24.0	48.0	48.0
700	15.6	15.6	23.4	23.4	46.8	46.8
750	15.2	15.2	22.8	22.8	45.6	45.6
800	14.8	14.8	22.2	22.2	44.4	44.4

Notes:

1. S_m = Stress intensity values per Table 2A of ASME II, Part D.
2. Alloy X S_m values are the lowest values for each of the candidate materials at corresponding temperature.
3. Stress classification per NB-3220.
4. P_m, P_L, P_b, Q, and P_e are defined in Table 3.1.10.

† Evaluation required for Design condition only.

†† Evaluation required for Levels A, B conditions only. P_e not applicable to vessels.

Table 3.1.8

LEVEL D: ALLOWABLE STRESS INTENSITY

Code: ASME NB
Material: Alloy X
Service Conditions: Level D (Accident)
Item: Stress Intensity

Temp. (Deg. F)	**Classification and Value (ksi)**		
	P_m	**P_L**	**$P_L + P_b$**
-20 to 100	48.0	72.0	72.0
200	48.0	72.0	72.0
300	46.3	69.45	69.45
400	44.6	66.9	66.9
500	42.0	63.0	63.0
600	39.6	59.4	59.4
650	38.4	57.6	57.6
700	37.4	56.1	56.1
750	36.5	54.8	54.8
800	35.5	53.25	53.25

Notes:

1. Level D stress intensities per ASME NB-3225 and Appendix F, Paragraph F-1331.
2. The average primary shear strength across a section loaded in pure shear may not exceed 0.42 S_u.
3. Stress classification per NB-3220.
4. P_m, P_L, and P_b are defined in Table 3.1.10.

Table 3.1.9

FRACTURE TOUGHNESS TEST REQUIREMENTS

Material	Test Requirement	Test Temperature	Acceptance Criterion
Ferritic steel with nominal section thickness of 5/8" or less	Not required per NF-2311(b)(1)	-	-
A516 Gr. 70 (greater than 5/8”) used for CEC base plate, CEC containment shell, CEC baffle plate, inlet plenum cover plate, inlet plenum corner gusset, inlet plenum side gusset, inlet plenum air-intake flange (see Note 2), MPC pedestal bearing pad, divider shell flange, divider shell flange gusset, lower and upper MPC guides, closure lid shear ring, closure lid strongback, closure lid bottom plate, closure lid outer shell, closure lid cover plate.	Not required per NF-2311 (b)(7)		
Weld material	Test per NF-2430 if: (1) either of the base materials of the production weld requires impact testing, or; (2) either of the base materials is A516 Gr. 70 with nominal section thickness greater than 5/8”.	See Note 1	Per NF-2330

Note:
1. Required NDT temperature = -40 deg. F for all materials in the HI-STORM UMAX VVM “NF” materials.
2. The optional A105/SA350 LF2 materials are exempted from impact testing per NF-2311(b)(7).

Table 3.1.10

ORIGIN, TYPE AND SIGNIFICANCE OF STRESSES IN THE HI-STORM UMAX SYSTEM

Symbol	Description	Notes
P_m	Primary membrane stress	Excludes effects of discontinuities and concentrations. Produced by pressure and mechanical loads. Primary membrane stress develops in the MPC Enclosure Vessel shell. Limits on P_m exist for normal (Level A), off-normal (Level B), and accident (Level D) service conditions.
P_L	Local membrane stress	Considers effects of discontinuities but not concentrations. Produced by pressure and mechanical loads, including earthquake inertial effects. P_L develops in the MPC Enclosure Vessel wall due to impact between the VVM guides and the MPC (near the top of the MPC) under an earthquake (Level D condition). However, because there is no Code limit on P_L under Level D event, a limit on the local strain consistent with the approach in the HI-STORM FW docket is used.
P_b	Primary bending stress	Component of primary stress proportional to the distance from the centroid of a solid section. Excludes the effects of discontinuities and concentrations. Produced by pressure and mechanical loads, including earthquake inertial effects. Primary bending stress develops in the top lid and baseplate of the MPC, which is a pressurized vessel. Lifting of the loaded MPC using the so-called "lift cleats" also produces primary bending stress in the MPC lid. Similarly, the top lid of the HI-STORM UMAX module, a plate-type structure, withstands the snow load or pressure loading from explosion by developing primary bending stress.
P_e	Secondary expansion stress	Stresses that result from the constraint of free-end displacement. Considers effects of discontinuities but not local stress concentration (not applicable to vessels). It is shown that there is no interference between component parts due to free thermal expansion. Therefore, P_e does not develop within any HI-STORM UMAX component.
Q	Secondary membrane plus bending stress	Self-equilibrating stress necessary to satisfy continuity of structure. Occurs at gross structural discontinuities. Can be caused by pressure, mechanical loads, or differential thermal expansion. The junction of MPC shell with the baseplate and top lid locations of gross structural discontinuity, where secondary stresses develop as a result of internal pressure. Secondary stresses would also develop at the two extremities of the MPC shell if a thermal gradient were to exist. However, because the top and bottom regions of the MPC cavity also serve as the top and bottom plenums, respectively, for the recirculating helium, the temperature field in the regions of gross discontinuity is essentially uniform, and as a result, the thermal stress adder is insignificant and neglected .
F	Peak stress	Increment added to primary or secondary stress by a concentration (notch), or, certain thermal stresses that may cause fatigue but not distortion. Because fatigue is not a credible source of failure in a passive system with gradual temperature changes, the cumulative damage factor from fatigue is not computed for HI-STORM UMAX components.

Table 3.1.11

MPC CONFINEMENT BOUNDARY STRESS INTENSITY LIMITS
FOR DIFFERENT LOADING CONDITIONS (ELASTIC ANALYSIS PER NB-3220)†

Stress Category	Design	Level A	Level D††
Primary Membrane, P_m	S_m	S_m	AMIN ($2.4S_m$, $.7S_u$)
Local Membrane, P_L	$1.5S_m$	$1.5S_m$	150% of P_m Limit
Membrane plus Primary Bending	$1.5S_m$	$1.5S_m$	150% of P_m Limit
Primary Membrane plus Primary Bending	$1.5S_m$	N/A	150% of P_m Limit
Membrane plus Primary Bending plus Secondary	N/A	$3S_m$	N/A
Average Shear Stress††††	$0.6S_m$	$0.6S_m$	$0.42S_u$

† Stress combinations including F (peak stress) apply to fatigue evaluations only.
†† Governed by Appendix F, Paragraph F-1331 of the ASME Code, Section III.
†††† Governed by NB-3227.2 or F-1331.1(d).

Table 3.1.12

STRESS AND ACCEPTANCE LIMITS FOR DIFFERENT
LOADING CONDITIONS FOR THE STEEL STRUCTURE OF THE
HI-STORM UMAX

STRESS CATEGORY	DESIGN + NORMAL	OFF-NORMAL	ACCIDENT†
Primary Membrane, P_m	S	1.33·S	See footnote
Primary Membrane, P_m, plus Primary Bending, P_b	1.5·S	1.995·S	See footnote
Shear Stress (Average)	0.6·S	0.6·S	See footnote

Definitions:

S = Allowable Stress Value for Table 1A, ASME Section II, Part D.
S_m = Allowable Stress Intensity Value from Table 2A, ASME Section II, Part D
S_u = Ultimate Stress

† Under accident conditions, the cask must maintain its physical integrity, the loss of solid shielding (concrete, steel, as applicable) shall be minimal and the MPC must remain retrievable.

Table 3.1.13

KEY INPUT DATA FOR FINITE ELEMENT MODEL OF HI-STORM UMAX VVM

Item	Value
Overall height of HI-STORM UMAX VVM (including Closure lid) from bottom of SFP to top of ISFSI pad	[PROPRIETARY INFORMATION WITHHELD IN ACCORDANCE WITH 10 CFR 2.390]
Height of CEC shell cavity including the top surface of the flange	[PROPRIETARY INFORMATION WITHHELD IN ACCORDANCE WITH 10 CFR 2.390]
Height of top lid above top of grade	[PROPRIETARY INFORMATION WITHHELD IN ACCORDANCE WITH 10 CFR 2.390]
Inside diameter of HI-STORM UMAX storage cavity (CEC shell) (See Note 1)	[PROPRIETARY INFORMATION WITHHELD IN ACCORDANCE WITH 10 CFR 2.390]
Outside diameter of Closure Lid	[PROPRIETARY INFORMATION WITHHELD IN ACCORDANCE WITH 10 CFR 2.390]
CEC shell thickness	[PROPRIETARY INFORMATION WITHHELD IN ACCORDANCE WITH 10 CFR 2.390]
Thickness of CEC flange	[PROPRIETARY INFORMATION WITHHELD IN ACCORDANCE WITH 10 CFR 2.390]
Lifting rib (in the Closure Lid outlet pipe) thickness	[PROPRIETARY INFORMATION WITHHELD IN ACCORDANCE WITH 10 CFR 2.390]
CEC Baseplate thickness	[PROPRIETARY INFORMATION WITHHELD IN ACCORDANCE WITH 10 CFR 2.390]
Material of construction	[PROPRIETARY INFORMATION WITHHELD IN ACCORDANCE WITH 10 CFR 2.390]
Ref. temperature for material properties	[PROPRIETARY INFORMATION WITHHELD IN ACCORDANCE WITH 10 CFR 2.390]
Concrete density(reference) †	[PROPRIETARY INFORMATION WITHHELD IN ACCORDANCE WITH 10 CFR 2.390]

Notes

1. UMAX System may have an MPC with increased shell thickness with corresponding increased MPC outside diameter and UMAX storage cavity inside diameter. Such systems are equivalent to or bounded by the existing structural analysis for the UMAX VVM listed in this table.

† The ISFSI Pad may have a dry density of 135 lb/ft,^{3} and the SFP may have a dry density of 120 lb/ft^3; however, using 150 lb/ft^3 as a reference dry density for all of the concrete is conservative.

Table 3.1.14

VALUES OF "K" AND "n" USED TO MODEL ELASTIC-PLASTIC BEHAVIOR OF HI-STORM UMAX COMPONENTS IN LS-DYNA

Component	Material	Ref. Temperature	K† (psi)	n†
MPC Lid	Alloy X	500ºF	1.055×10^5	0.235
MPC Shell	Alloy X	475ºF	1.156×10^5	0.246
MPC Baseplate	Alloy X	350ºF	1.161×10^5	0.236
CEC Shell	A516 Gr. 70	150ºF	1.124×10^5	0.171
Fuel Basket	Metamic-HT	644ºF	1.764×10^4	0.060
		617ºF	1.879×10^4	0.056
		572ºF	2.116×10^4	0.051
		392ºF	2.712×10^4	0.075

† K and n are defined in Subsection 3.1.3.1.

Table 3.1.15

KEY INPUT DATA FOR LS-DYNA MODEL OF GOVERNING MPC ENCLOSURE VESSEL

Item	Value
Overall Height of MPC	213 in (for maximum length fuel)
Outside diameter of MPC	75.5 in (see Notes 1 and 2)
MPC upper lid thickness	4.5 in
MPC lower lid thickness	4.5 in
MPC shell thickness (see Note 2)	0.5 in
MPC baseplate thickness	3.0 in
Material	Austenitic stainless steel (Alloy X in Docket # 72-1014)
Ref. temperature for material properties	[3.1.7]

Notes:

1. Input value is slightly less than the outside diameter (75.75 in) of the MPC-37 and MPC-89 per the Licensing drawings in Section 1.5. The use of a smaller diameter is conservative because it increases the clearance gap between the MPC and the MPC guide plates, which results in higher lateral impact loads between them during the Design Basis Earthquake.

2. UMAX System may have an MPC with increased shell thickness with corresponding increased MPC outside diameter and UMAX Divider and Container Shell inner diameters. Such systems are bounded by the existing structural analysis for the governing MPC listed in this table because of the increased structural capacity of the thicker shell in addition to the reasons provided in Note 1 above.

3.2 WEIGHTS AND CENTERS OF GRAVITY

Table 3.2.1 provides bounding weight data of the movable components (Closure lid, MPC, HI-TRAC and the transporter) required for the structural analysis of the HI-STORM UMAX. The weight data is selected to bound all types of MPCs, HI-TRACs and transporters that may be employed with the HI-STORM UMAX system.

Table 3.2.2 provides limiting (maximum and minimum) dimensional data that bound the MPC types certified in docket number 72-1032.

Because the HI-STORM UMAX is immovable and is situated underground, its CG data is not germane to safety evaluation. The Closure Lid is essentially a radially symmetric structure and, as such, its center-of-gravity is closely aligned with its axis of symmetry.

Table 3.2.1

HI-STORM MPC, LID, HI-TRAC AND TRANSPORTER WEIGHT & PRESSURE DATA USED IN STRUCTURAL ANALYSES

Item	Bounding Weight or Value
Closure Lid	35,000 lb
Loaded Transfer cask	270,000 lb (Note 1)
MPC	110,000 lb (Note 1)
LOADED TRANSPORTER DATA (Note 2)	
• Weight of empty Transporter plus rigging (used to carry the loaded transfer cask)	180,000 lbs
• Reference length and width of each load patch (two load patches per transporter)	197.1875 inch long × 29.5 inch wide (Note 2)
• Computed average normal pressure (based on loaded transporter weight and calculated area of two load patches)	38.68 psi

Notes:

1. Maximum MPC weight and HI-TRAC weights are intended to bound all MPCs (listed in Table 1.2.1) and transfer casks that may be used to load fuel in the HI-STORM UMAX VVM.
2. Transporter reference data is based on typical transporters being used at Holtec ISFSI sites.

Table 3.2.2

LIMITING PARAMETERS

	Item	PWR	BWR
1.	Minimum unirradiated fuel assembly length (nominal), inch	157	171
2.	Maximum unirradiated fuel assembly length (nominal), inch	199.2	181.5[1]
3.	Maximum MPC length, inch	213	195
4.	Minimum MPC length, inch	171	185

[1] Maximum fuel assembly length for the BWR fuel assembly refers to the maximum fuel assembly length plus an additional 5” to account for a Damage Fuel Container (DFC).

3.3 MECHANICAL PROPERTIES OF MATERIALS

This section provides the mechanical properties used in the structural evaluation of the HI-STORM UMAX VVM. The properties include yield stress, ultimate stress, modulus of elasticity, Poisson's ratio, weight density, and coefficient of thermal expansion. Values are presented for a range of temperatures which envelope the maximum and minimum temperatures under all service conditions applicable to the HI-STORM UMAX system components.

The materials selected for use in the HI-STORM UMAX VVM are presented on the drawings in Section 1.5. In this chapter, the materials are divided into two categories, structural and nonstructural. Structural materials are materials that act as load bearing members and are, therefore, significant in the stress evaluations. Materials that do not support mechanical loads are considered nonstructural. For nonstructural materials, the principal property that is used in the structural analysis is weight density. In local deformation analysis, however, such as the study of penetration from a tornado-borne missile, the properties of plain concrete in the HI-STORM Closure Lid are included.

Table 2.6.2 lists applicable codes, materials of construction, and ITS designations for the functional parts in the HI-STORM UMAX system.

3.3.1 Structural Materials

Tables 3.3.1 and 3.3.2 provide the numerical values of the material properties needed for structural analysis.

a. Reinforced concrete is used in the construction of the concrete ISFSI Structures, namely, the ISFSI pad, the SFP, and possibly the optional Enclosure Wall. All reinforced concrete load bearing structures in the HI-STORM UMAX ISFSI will conform to stress criteria of ACI-318(2005). Table 3.3.4 provides the required properties for reinforced concrete.

b. Weld materials: All weld materials utilized in the welding of the Code components comply with the provisions of the appropriate ASME subsection (e.g., Subsection NB for the MPC enclosure vessel) and Section IX. All non-code welds will be made using weld procedures that meet Section IX of the ASME Code. The minimum tensile strength of the weld wire and filler material (where applicable) will be equal to or greater than the tensile strength of the base metal listed in the ASME Code.

c. Subgrade material: As described in subsection 1.2.2, the subgrade material in Space A in Figure 2.4.4 serves a structural function under certain design basis loading events, namely the Design Basis Earthquake (DBE) and the Design Basis Missile (DBM) impact events. Therefore, it is classified as ITS Category C material. The critical characteristics assumed for qualifying the "UMAX" ISFSI under the above mentioned design basis loadings are summarized in Table 3.3.4.

3.3.2 Nonstructural Materials

Plain concrete used in the Closure lid and the non-organic insulation used on the outside surface of the Divider shell are the two non-structural materials used in the HI-STORM UMAX VVM assembly.

The primary function of the unreinforced concrete in the VVM Closure Lid is shielding. Unreinforced concrete is not considered as a primary load-bearing (structural) member. However, its ability to withstand compressive, bearing and penetrant loads under the design basis and various service conditions is analyzed. Chapters 2 and Chapter 8 provide requirements on plain (unreinforced) concrete. The compressive strength, bearing stress limit and the reference dry density applicable to the Closure Lid plain concrete is specified in Table 3.3.3 herein. The allowable bearing strength of plain concrete for normal loading conditions is calculated in accordance with ACI-318 (2005) [3.4.6]. The procedure specified in ASTM C-39 [3.3.1] is utilized to verify that the assumed compressive strength will be realized in the actual in-situ pours.

3.3.3 ISFSI Materials

The mechanical properties of reinforced concrete, subgrade and undergrade required for stress and strength analysis are provided in Tables 3.3.4 and 3.3.5.

Table 3.3.1A

RELEVANT MATERIAL PROPERTIES FOR HI-STORM UMAX
YIELD, ULTIMATE, LINEAR THERMAL EXPANSION,
YOUNG'S MODULUS

Temp. (Deg. F)	**A516 and A515, Grade 70**			
	S_y	S_u	α	**E**
-40	38.0	70.0	---	29.98
100	38.0	70.0	6.5	29.26
150	35.7	70.0	6.6	29.03
200	34.8	70.0	6.7	28.8
250	34.2	70.0	6.8	28.55
300	33.6	70.0	6.9	28.3
350	33.05	70.0	7.0	28.1
400	32.5	70.0	7.1	27.9
450	31.75	70.0	7.2	27.6
500	31.0	70.0	7.3	27.3
550	30.05	70.0	7.3	26.9
600	29.1	70.0	7.4	26.5
650	28.2	70.0	7.5	26.0
700	27.2	70.0	7.6	25.5
750	26.3	69.1	7.7	24.85
800	25.5	64.3	7.8	24.2

Table 3.3.1B

RELEVANT MATERIAL PROPERTIES FOR HI-STORM UMAX
YIELD, ULTIMATE, LINEAR THERMAL EXPANSION,
YOUNG'S MODULUS

Temp. (Deg. F)	A240-304, A240-304L, A479-304 and A479-304L (Note 1)			
	S_y	S_u	α	E
-40	25.0	70.0	---	28.88
100	25.0	70.0	8.6	28.12
200	21.4	66.1	8.9	27.5
300	19.2	61.2	9.2	27.0
400	17.5	58.7	9.5	26.4
500	16.4	57.5	9.7	25.9
600	15.5	56.9	9.9	25.3
650	15.2	56.7	9.9	25.05
700	15.0	56.4	10.0	24.8
750	14.7	56.0	10.0	24.45
800	14.5	55.4	10.1	24.1

Notes:

1. The yield and ultimate strength values are the lowest of the values for the materials (A240-304, A240-304L, A479-304 and A479-304L) at corresponding temperature; the material properties are obtained from [3.3.2].

Table 3.3.2

A36 MATERIAL PROPERTIES

Temp. (Deg. F)	**A36**			
	$\mathbf{S_y}$	$\mathbf{S_u}$	$\boldsymbol{\alpha}$	**E**
-40	36.0	58.0	---	29.98
100	36.0	58.0	6.5	29.26
150	33.8	58.0	6.6	29.03
200	33.0	58.0	6.7	28.8
250	32.4	58.0	6.8	28.55
300	31.8	58.0	6.9	28.3
350	31.3	58.0	7.0	28.1
400	30.8	58.0	7.1	27.9
450	30.05	58.0	7.2	27.6
500	29.3	58.0	7.3	27.3
550	28.45	58.0	7.3	26.9
600	27.6	58.0	7.4	26.5
650	26.7	58.0	7.5	26.0
700	25.8	58.0	7.6	25.5

Definitions:

S_y = Yield Stress (ksi)
α = Mean Coefficient of thermal expansion (in./in./^{0}F x 10^{-6})
S_u = Ultimate Stress (ksi)
E = Young's Modulus (psi x 10^6)

Table 3.3.3

HI-STORM UMAX VVM LID PLAIN CONCRETE PROPERTIES

Input Parameter	Value
Density, lbf/ft^3	Table 2.3.2
Poisson's ratio	0.17
Compressive strength, psi	Table 2.3.2
Concrete Allowable Bearing Stress, psi	4,420 (Note 1)
Young's modulus, psi	57,000 × (Concrete compressive strength in psi)$^{1/2}$

Notes:
1. Per ACI-318 (2005), Sec. 10.17.1 and Sec. 9.3.2.4. Since the plain concrete in the HI-STORM UMAX VVM lid is always confined, the allowable value is doubled.

Table 3.3.4

REFERENCE AND DERIVED PROPERTIES OF ISFSI REINFORCED CONCRETE, SUBGRADE, AND UNDERGRADE

Property	Value
Concrete and SES Compressive Strengths (psi)	Table 2.3.2
Concrete Rupture Strength (psi)	335.4
SES Rupture Strength (psi)	158.1
Concrete Allowable Bearing Stress (psi)	2486.3
SES Allowable Bearing Stress (psi)	552.5
Concrete Mean Coefficient of Thermal Expansion (in/in-deg. F)	5.5E-06
Concrete Modulus of Elasticity (psi)	57,000 × (Concrete compressive strength in psi)$^{1/2}$
Concrete Reference Dry Density	Table 2.3.2
Subgrade and Undergrade Strain Compatible Modulus of Elasticity (ksi) (see Figure 2.4.4 for identification of subgrade spaces)	Subgrade Space A (i.e., SES): 102 Subgrade Space B: 14.0 Undergrade Spaces C and D: 17.7

Table 3.3.5

SOIL PROPERTIES, COMPUTED SETTLEMENT, AND CORRESPONDING ELASTIC MODULUS FOR THE UNDERGRADE

	Item	**Value**
1.	Water Content ‘w_n’ Soil Parameter ‘a’ Soil Parameter ‘b’ Poisson’s Ratio	14% 0.18 0.13 0.45
2.	Derived Properties for the Undergrade (Notes 1 and 3): Computed Long-Term Settlement (in) (Note 2) Computed Elastic Modulus (psi)	 < 0.12 6,230
3.	Values used in the Structural Analyses Model for Undergrade: Limiting Long-Term Settlement (in) Corresponding Elastic Modulus (psi)	 From Table 2.3.2 3,200

Note 1: The substrate characteristics are obtained using the density data from Table 2-3 and Table 5-1 of reference [3.3.3]. The soil compaction index ‘Cc’ is a direct function of soil parameters w_n, a, and b per [3.3.3]. The long-term settlement and the elastic modulus are derived using the relationships in [3.3.4].

Note 2: See Table 2.3.2 for the values of settlement (greater than those computed here for conservatism) used as the Design Basis data for qualification of the ISFSI structures.

Note 3: The Design Basis settlement has been set at a higher value than that computed for the SFP to allow for the variation in the soil parameters at a host site.

3.4 GENERAL STANDARDS FOR CASKS

3.4.1 Chemical and Galvanic Reactions

The potential of chemical and galvanic action in the HI-STORM UMAX VVM assembly is evaluated in Chapter 8 of this FSAR.

3.4.2 Positive Closure

The confinement boundary of the HI-STORM UMAX system is seal welded in its entirety. The only access to the MPC storage cavity is through the VVM Closure Lid, which weighs well over 16 tons and must be lifted vertically a substantial distance before it can be separated from the VVM body. Furthermore, the removal of the lid requires the use of a special lifting device. Thus inadvertent opening of the VVM cavity is not feasible or credible.

3.4.3 Lifting Devices

3.4.3.1 Identification of Lifting Devices and Required Safety Factors

The only component in the HI-STORM UMAX system requiring lifting and handling is the Closure Lid. The Closure Lid is equipped with a four point lift system that meets the stress requirement of ANSI N14.6 [3.4.3] and Regulatory Guide 3.61 [1.0.4].

As required by Reg. Guide 3.61, lifting operations applicable to the VVM Closure lid are analyzed. Because of the nature of the HI-STORM UMAX system, lid placement or removal may occur with loaded MPCs inside the VVM cavity. Therefore, a stress analysis to demonstrate compliance with ANSI N14.6 to provide the assurance that a structural failure will not occur during lifting is summarized in this chapter.

The governing requirement for the lifting component itself (the lift lug plates) is that they must meet the primary stress limits prescribed by ANSI N14.6-1993 [3.4.3]; the welds in the load path, near the lifting holes, are required to meet the condition that stresses remain below yield under three times the lifted load (per Reg. Guide 3.61). Further, for additional conservatism, away from the lifting location, the ASME Code limit for the Level A service condition applies. Only the lifting component itself is the "significant-to-handling" (STH) part.

The lifting analysis is performed by conservatively assuming that the dead load is amplified by 15%.

3.4.3.2 Analysis of Lifting Scenarios- HI-STORM UMAX VVM Closure Lid Lifting Analysis (Load Case 04 in Table 2.4.1)

The Closure Lid is an axi-symmetric plate-type structure filled with concrete to provide radiation shielding. As shown in the Licensing Drawings, three lift lug plates with four orthogonally situated lift holes are welded together to lift and handle the Closure Lid. The diameter of the Closure Lid is much larger than the CEC shell I.D. which eliminates the risk of the lid falling into the cavity and striking the MPC stored below. Nevertheless, the Closure Lid lifting points and the Lid itself are required herein to meet the most stringent stress criteria from the regulatory literature, as follows:

1. The primary stresses at the lifting points must meet the limits in ANSI N14.6 with the dead load amplified by 15% to incorporate dynamic effects. The allowable stress per ANSI N14.6 is lesser of 1/10th of the ultimate strength or 1/6th of the material Yield Strength.

2. The average stress in the welds joining the lift lug plates for Closure Lid handling under the amplified dead load must meet the limit set down in Regulatory Guide 3.61, which limits the stress to 1/3rd of the material's Yield Strength.

3. The primary stresses in the lid structure under the amplified dead load must meet Level A stress limits in the ASME Code, Subsection "NF" for class 3 linear structures.

Because of the simplicity of the Closure Lid configuration, the lifting analysis is performed using strength of material approach. The details of the calculations are presented in the calculation package [3.4.1] supporting this FSAR. Lifting slings that attach to the lugs shall be sized to meet the safety factors set forth in ANSI B30.3 [3.4.2].

Table 3.4.1 summarizes key results obtained from the lifting analyses for the limiting HI-STORM UMAX VVM Closure Lid design (i.e., the stainless steel lid design) for a bounding set of input design loads.

It is concluded that all structural integrity requirements are met during a lift of the HI-STORM UMAX VVM Closure Lid. All factors of safety are greater than 1.0.

3.4.3.3 Safety Evaluation of Lifting Scenarios

As can be seen from the above, the computed factors of safety have a large margin over the allowable (of 1.0) in every case. In the actual fabricated hardware, the factors of safety will likely be much greater because of the fact that the actual material strength properties are generally substantially greater than the Code minimums. Minor variations in manufacturing, on the other hand, may result in a small subtraction from the above computed factors of safety. A 10CFR72.48 safety evaluation will be required if the cumulative effect of manufacturing deviation and use of the CMTR (or CoC) material strength in a manufactured hardware renders a factor of safety to fall below the above computed value. Otherwise, a 10CFR72.48 evaluation is not necessary. The above criterion applies to all lift calculations covered in this FSAR.

3.4.4 Heat

The evaluation of the HI-STORM UMAX system under thermally significant conditions listed in Section 2.5 is reported in Chapter 4.

a. Summary of Pressures and Temperatures

Required input data for the structural analysis of the HI-STORM UMAX VVM assembly is contained in Tables 2.3.1, 2.3.2 and the Licensing Drawings.

b. Differential Thermal Expansion

i. Normal Hot Environment

All clearances between the MPC and the HI-STORM UMAX VVM are considerably larger than the thermal expansion that may occur during system operations. Therefore, no interferences between the MPC and the HI-STORM UMAX VVM will occur due to thermal expansion of the loaded MPC. The interfaces that may potentially be subject to interference and their reference dimensions from the Licensing Drawings are provided below:

Interfacing Parts	Engineered Gap in Inches (shown in Licensing Drawings)
MPC-to-Divider Shell (diametral clearance)	10.25
MPC-to-Closure Lid (vertical clearance)	23.5
MPC-to-MPC Top Guide Plates (diametral clearance)	0.5
MPC-to-MPC Bottom Guides (diametral clearance)	0.5
Closure Lid-to-CEC Flange (diametral clearance)	1.0

Simplified calculations summarized in Section 4.4 (Chapter 4) show that the engineered gap in the system for each of the above listed interfaces is an order of magnitude (or more) greater than its reduction from thermal expansion under the operating condition corresponding to the maximum temperature differential scenario between the mating parts that form the interface.

ii. Design Basis Fire Event (Load Case 05 in Table 2.4.1)

The thermal analysis of the Design Basis Fire event is documented in Chapter 4 and evaluated in Chapter 12 for its safety consequence. It is shown in Chapter 4 that the fire accident has a small effect on the MPC temperatures because of blocking action of the underground storage system and the short duration of the fire event. Therefore, a structural evaluation of the MPC under the postulated fire event is not required for HI-STORM UMAX.

Likewise, it is shown that the load bearing components in the HI-STORM UMAX assembly will remain well below the temperature that may induce a significant structural deformation or collapse.

3.4.4.1 Safety Analysis

3.4.4.1.1 Design Basis Flood (Load Case 07 in Table 2.4.1)

Unlike free standing casks, moving flood water is not an event of safety consequence to HI-STORM UMAX: The buried configuration of the HI-STORM UMAX system renders it immune from sliding (that is germane to above ground freestanding casks) under the action of a design basis flood.

3.4.4.1.2 Design Basis Earthquake (Load Case 03 in Table 2.4.1)

The treatment of Load Case 03 (Design Basis Earthquake) for the HI-STORM UMAX system, including the model and analysis methodology, is provided below. The parallel analysis for the structurally strengthened design of "UMAX," termed Version MSE in Subsection 1.0.2, under the Most Severe Earthquake event (Table 2.3.10) is provided at the end of this sub-paragraph.

The HI-STORM UMAX system, plus its contents, may be subject to the Design Basis Earthquake (DBE) defined by the response spectra in Figures 2.4.1 and 2.4.2. As explained in Chapter 2, the DBE has been defined for the HI-STORM UMAX ISFSI to ensure that the operative spectra (Figures 2.4.1 and 2.4.2) essentially envelope the corresponding site DBE spectra at virtually all US sites. Because the VVM is buried in the substrate, tip-over of the MPC is not possible.

Under the action of lateral seismic loads, the CEC Container Shell globally acts as a beam-like structure supported on a foundation driven by the site seismic accelerations. During a seismic event, the lateral loading on the CEC consists of:

2. Inertia force from CEC self-weight

3. Inertia forces from the Closure Lid self-weight

4. Interface forces from the rattling of the MPC within its confines of the CEC and the rattling of the contents inside the MPC

5. Interface forces from the subgrade and from the SFP

The CEC Container Shell may develop longitudinal stresses as it bends like a beam to resist the input seismic loads. In addition, the CEC Container Shell and the Divider Shell are subject to ovalizing action from the loads. Both effects are captured in the seismic analysis.

The seismic analysis consists of three discrete steps, namely:

A. Develop the Design Basis Seismic Model (DBSM) for the HI-STORM UMAX ISFSI and perform the Design Basis Earthquake Soil-Structure Interaction (SSI) analysis.

B. Seismic Qualification of the VVM Components based on the SSI analysis results.

C. Stress analysis of the ISFSI structures using the dynamic loads obtained from the LS-DYNA SSI analysis using DBSM.

A. Design Basis Seismic Model and SSI Analysis

As discussed in Section 2.4, based on the lower bound shear wave velocity profile of US nuclear power plants (Figure 2.4.3) and the input seismic acceleration time history derived from the Regulatory Guide 1.60 seismic response spectrum, a two-step soil seismic response analysis using the computer code SHAKE2000 [3.4.4] and LS-DYNA [3.4.5] is performed to establish a bounding seismic loading condition for the HI-STORM UMAX underground fuel storage system. The 1-D SHAKE analysis model consists of 21 native soil layers of the HI-STORM UMAX ISFSI site with a total thickness of 101 ft; the top of the 7^{th} soil layer is aligned with the bottom of the SFP. The total soil depth of the SSI Model is about five times the height of the underground ISFSI. Figure 3.4.1 shows the LS-DYNA soil model for the seismic response analysis. It is noted that the lateral dimension of the ISFSI soil model is significantly greater than that of the ISFSI. The periphery nodes of the soil model space at the same elevation are constrained to move together to simulate the seismic response of the semi-infinite space of soil. According to the numerical study on various lateral boundary conditions of the finite element soil model [2.4.1], this lateral boundary condition, also known as a "slave boundary condition", is appropriate to predict the soil response in a seismic event. The same soil model and input seismic motion used in the LS-DYNA soil seismic response analysis is used later in the DBSM developed to perform SSI analyses for the HI-STORM UMAX ISFSI.

The object of the DBSM is to obtain conservative values of the loads on the ISFSI structures under the Design Basis Earthquake, which is defined as the response spectra at both the ground surface and the ISFSI foundation elevations as shown in Figures 2.4.1 and 2.4.2 (obtained from the two-step SHAKE/LS-DYNA soil seismic response analysis). Since some of the HI-STORM UMAX steel members (i.e., the closure lid, the Divider Shell, and the CEC) have carbon steel and stainless steel design options, the heavier carbon steel design option is considered in the DBSM to yield conservative SSI analysis results. Following the approach used in the safety evaluation of HI-STORM 100U, the reference VVM assemblage used in the seismic qualification is a 5 by 5 array.

The actual array size, as noted in the HI-STORM 100U (docket number 72-1014) may be much larger. The essential attributes of the DBSM are:

1. The DBSM is developed using the finite element code LS-DYNA; this HI-STORM UMAX ISFSI soil-structure model consists of loaded VVMs, concrete pads, and soil spaces with properties as defined in Table 2.3.2. The HI-STORM UMAX VVM Model is discussed in detail in Paragraph 3.1.3.1.

2. The model has the capability to remove the lateral subgrade all the way down to the bottom of the SFP (which conservatively represents an excavated configuration during additional site construction such as to extend the ISFSI).

3. The model has the ability to simulate a loaded cask transporter arrayed on top of the ISFSI pad to enable the stability of the transporter and the structural margin of safety in the ISFSI pad to be determined.

4. The MPC is represented by a solid rigid cylinder of mass equal to its total mass. This means that all internal masses will move in unison and the inertia forces of the MPC are maximized, which will conservatively result in greater impact loads applied to MPC guides and the CEC base plate.

5. The ISFSI pad and SFP are simulated as a flexible plate-type structure represented by layered solid elements. Proper contact interfaces are defined among the ISFSI pad, the SFP, the ISFSI subgrade and undergrade. For conservatism, the optional enclosure wall is not considered in the DBSM.

6. The VCT, along with the carried transfer cask, is modeled as a freestanding rigid body.

7. The ISFSI pad is characterized by an LS-DYNA inelastic concrete model (MAT_PSEUDO_TENSOR) to account for energy dissipation in the concrete due to the impact loading from the loaded VCT. The SFP and soil are modeled using the LS-DYNA elastic material model. For the case where cracking of the concrete needs to be considered, the Young's Modulus of the SFP is reduced to 50% of its nominal value per the guidance in Section 3.4 of [3.4.15].

8. Proper element size and time step controls in the dynamic model are implemented following the guidance in references [3.4.14] and [3.4.15].

The previously described DBSM is used to perform SSI analyses for the four applicable HI-STORM UMAX ISFSI loading scenarios identified and listed in Table 3.4.2. Figures 3.4.2 and 3.4.3 show the corresponding LS-DYNA models of the two ISFSI configurations considered in the four SSI analyses. In each case, the Design Basis Earthquake is applied to the bottom of the LS-DYNA model using the acceleration time histories obtained from the previously described 1-D SHAKE seismic response analysis.

Table 3.4.3 lists the peak ISFSI interface loads obtained from the LS-DYNA SSI simulations of the four loading scenarios listed in Table 3.4.2. These dynamic peak interface loads will be used in the structural qualification of the ISFSI. In addition to the ISFSI interface loads, the SSI analyses also demonstrate that the exposed SES due to excavation for ISFSI expansion can remain intact during the DBE event without the lateral support from the optional enclosure wall. Figure 3.4.4 shows the maximum tensile stress experienced by the SES. The SES has a much greater tensile capacity (see Table 3.3.4) and the safety factor of the SES against rupture is calculated as

$$\begin{aligned} Safety\ Factor &= \frac{Rupture\ strength}{Maximum\ computed\ tensile\ stress\ \ from\ Figure\ 3.4.4} \\ &= \frac{158.1 psi}{77.91 psi} = 2.03 \end{aligned}$$

Finally, the stability of the loaded VCT is confirmed by the LS-DYNA SSI analyses.

B. <u>Seismic Qualification of VVM Components</u>

In addition to the SSI analysis, a governing MPC to MPC guide impact analysis is performed. Figure 3.4.5 shows the MPC to guide impact model, where the MPC guide is fixed at its base and the loaded MPC rotates about the bottom pivot point with an angular velocity to result in an impact force that significantly bounds, or more precisely is over 2 times, the load capacity of the MPC guide (see Table 3.4.4 and Figure 3.4.6a). The MPC enclosure vessel and its contents are explicitly modeled with sufficiently fine mesh density, following the conclusion obtained from the mesh sensitivity study performed for the MPC shell in Docket number 72-1014, to capture the high stress gradient at the impact location. Results obtained from the MPC-to-guide impact analysis and those from the above mentioned SSI analyses are used to structurally qualify HI-STORM VVM components herein.

The CEC Components and parts of the MPC subject to significant loadings during the DBE event are:

a. Divider and CEC shell (subject to ovalization)
b. MPC shell (bending of the shell as a beam, resulting in axial membrane stress in the shell)
c. MPC top and bottom guides (subject to buckling)
d. Lateral loading on the fuel basket panels (must meet the g-load limit).
e. Localized strain in the MPC shell (due to impact of the MPC with the MPC guides if excessive, may cause breach of confinement)

The focus of safety analysis of each component under the DBE event is somewhat different as summarized below:

a. CEC and Divider Shell: These shells are subject to ovalizing forces which may, in theory, cause them to deform and impede future retrievability of the loaded MPC. Since the SES between the ISFSI pad and the SFP is much stiffer than the subgrade material in the HI-STORM 100U design (the SES modulus of elasticity is at least 6.76 times the value of 100U subgrade based on the required shear wave velocity values), ovalization is not a credible concern for HI-STORM UMAX VVM given the very high safety factor (>18) in the 100U design.

b. Primary stress in the MPC shell: The maximum stress intensity in the MPC shell is compared against the allowable stress intensity for the Level D condition. The safety factor is computed as the ratio of the allowable stress intensity under Level D service condition for "NB" components to the maximum computed longitudinal flexural stress intensity in the MPC shell. Figure 3.4.6b shows the distribution of maximum shear stress (i.e., ½ of the stress intensity) in the MPC shell under the bounding impact condition. As documented in the calculation package [3.4.1], the safety factor of the MPC shell against primary stress is well above 1.0.

c. Top and Bottom MPC Guides: The MPC guides are subject to in-plane rattling loads from the lateral movement of the MPC under the inertia loading from the earthquake. The maximum in-plane load bearing capacity of the longest MPC guide permitted by the system design (see Licensing Drawings) is computed and compared with the maximum dynamic loads obtained from the LS-DYNA analysis. The safety factor is calculated as the ratio of the in-plane load bearing capacity and the actual maximum load computed by the LS-DYNA analysis. The analyses summarized in the Calculation Package [3.4.1] shows that the minimum safety factor from the array of dynamic simulations is greater than 1.0.

d. Loading on the Fuel Basket panel: The minimum lateral g-load to which all of the MPCs listed in Table 1.2.1 have been qualified is shown in Table 3.1.1 (corresponding to the non-mechanistic tip-over event). As shown in Figure 3.4.7, the maximum fuel g-load predicted by the LS-DYNA simulation, conservatively measured at the top lid of the MPC, is significantly smaller than the minimum permitted value for the array of MPC types that may be stored in the HI-STORM UMAX system. Although the calculated fuel basket peak temperature under the HI-STORM UMAX normal storage condition with wind (see Table 4.4.15) is slightly higher (32 ºC) than the fuel basket temperature considered in the MPC to MPC guide impact analysis, which leads to approximately 12% reduction in the yield strength of the fuel basket material, the much smaller MPC impact loading (less than half of the acceptable loading) ensures that the structural integrity of the fuel basket in the HI-STORM UMAX system is less challenged compared with the that in the above ground HI-STORM systems.

e. Maximum Local Strain in the Confinement Boundary in the Impact Region: The MPCs are constrained from uncontrolled lateral motion by the radial guide plates located at their baseplate and top lid elevations. Even a small clearance between the MPCs and the

MPC guides can lead to a high localized strain in the region of the shell where the impact from rattling of the canister under a seismic event occurs. Based on the parametric study performed from HI-STORM 100 U, the extent of local strain from impact is minimized by locating the MPC top guide in the vertical direction such that the mid-height of the impact footprint is aligned with the lower surface of the MPC lid in the HI-STORM UMAX design. Thus the location of impact is removed from the lid-to-shell weld junction. Figure 3.4.8 shows the maximum MPC shell maximum plastic (true) strain obtained for the bounding MPC-to-guide impact simulation. The plastic strain developed in the MPC shell is only a small fraction of the acceptable value based on the very conservative recommendation in [3.1.3]. Therefore, the integrity of the confinement boundary is assured.

Table 3.4.4 summarizes the seismic qualification analysis results for VVM components.

To reduce the occupational and site boundary dose rates, an optional divider shell shield ring (attached to the divider shell) can be implemented. The drawings provided in Chapter 1 give more information about this shield ring specification. The optional divider shell shield ring has no adverse effect on the structural performance of the VVM. The attachment welds between the optional ring and the divider shell have been sized appropriately so that they can support the amplified self-weight of the optional shield ring under seismic loading.

C. Strength Qualification of the ISFSI Structure

Under the Design Basis Earthquake (Figures 2.4.1 and 2.4.2), the loads exerted on the Support Foundation Pad and the ISFSI Pad are obtained from the LS-DYNA SSI simulations listed in Table 3.4.2. Table 3.4.3 lists the peak ISFSI interface loads obtained from various LS-DYNA runs listed in Table 3.4.2. In order to incorporate an additional margin of safety in the ISFSI structural analysis, these unfiltered dynamic bounding interface loads are directly used for the structural evaluation of ISFSI components. The use of the bounding loads is in keeping with a similarly bounding value of settlement specified for the strength analysis of the SFP and the ISFSI pad (see Table 2.3.2).

The SFP and ISFSI pad are required to meet the minimum structural requirements set down in Table 2.3.2 and the Licensing Drawings. The ISFSI pad is required to satisfy ACI-318 (2005) [3.4.6] strength limits under all applicable load combinations (Table 2.4.3).

Table 2.4.3 specifies the load combinations used in the strength analysis of the ISFSI structures. The following discrete analyses are required:

(i) Compute the long-term settlement of the undergrade supporting the SFP assuming all VVM locations are loaded for the entire Design Life: Determine the "effective" elastic modulus of the subgrade under the SFP to simulate the effect of settlement in the structural analyses model. As discussed previously, the long-term settlement of the undergrade from the loaded VVMs and the dead weight of the SFP are very small because the combined equivalent density of the loaded VVM's and the SFP is nearly equal to the density of the excavated

subgrade.

This methodology of settlement computation is based on classical soil mechanics and is summarized below.

1. Compute the total long-term settlement, "d", of the subgrade under the SFP (Space C) over the Design Life assuming that the total load "P" (modeled as a uniform pressure at the top of the subgrade) is equivalent to that produced by the SFP fully populated with loaded VVMs for the entire life using the methodology in [3.4.7].

2. Determine an "effective" elastic spring constant "K" of Space C that emulates the cumulative settlement:

$$K = P/d.$$

3. Using the spring constant computed above, which accounts for the effect of long-term settlement under static loading, an appropriate elastic modulus is defined for the soil column under the SFP. The degraded soil modulus so defined is used in the finite element model of the SFP to evaluate the pad flexure under the factored dead load.

The maximum permitted settlement of the subgrade below SFP is limited to the value specified in Table 2.3.2. If the Table 2.3.2 limit cannot be met, remedial measures such as pilings must be used, unless a site-specific analysis is performed using the same methodology described in Paragraph 3.4.4.1 to qualify the ISFSI structure for the increased settlement.

(ii) Prepare a finite element model of the pads in ANSYS and determine the stress field under the factored Dead and Live loads with the settlement based "degraded" elastic moduli.

(iii) Compute the stress field in the pads under factored seismic loads using dynamic elastic modulus corresponding to the minimum shear wave velocity of the subgrade specified in Table 2.3.2.

(iv) Use the bounding peak loads obtained from the dynamic analysis to compute the stress fields in the pads (SFP and ISFSI pad).

(v) Combine the factored loads and determine the total stress resultants. Compare with the respective section strengths to establish the factors of safety for the SFP and the ISFSI pad.

(vi) Compute the bearing stress (or load) on the subgrade under the ISFSI pad using the combined factored loads from the transporter and the ISFSI pad and compare with the corresponding allowable limit to establish the safety factor for the subgrade under the ISFSI pad.

A summary of the analyses and the associated margins of safety are discussed below:

The structural evaluation of the HI-STORM UMAX ISFSI is performed using the commercial computer code ANSYS [3.1.1]. The constituents of the ISFSI, namely the Support Foundation Pad (SFP), the subgrade under the support foundation pad (the undergrade), the ISFSI pad and the subgrade lateral to the CEC under the ISFSI pad are all modeled using linear elastic SOLID45 elements. The element mesh is intentionally kept fine in the areas of load application on the SFP and the ISFSI pad. The lateral subgrade adjacent to the HI-STORM UMAX ISFSI (Spaces B and D in Figure 2.4.4) is included in the FE model and extends laterally a distance that exceeds the overall depth of the FE model considered for structural analysis. For convenience of load application, the footprint of the CEC base on the SFP is carefully articulated in the finite element model. The substrate under the SFP is terminated at approximately 101.0 ft below the ISFSI pad, which is consistent with the Design Basis Seismic Model discussed previously. The “base” model (Simulation Model I) considers that all the storage locations in ISFSI are populated and experience identical peak vertical seismic loading from Table 3.4.3, which bounds the peak result obtained from the LS-DYNA SSI solution as discussed previously. Because of the symmetric geometry and loading, a quarter symmetric finite element model is sufficient to represent the fully loaded ISFSI. Figure 3.4.9 shows the finite element model of HI-STORM UMAX ISFSI. For a non-symmetric model as in case of Simulation Model II, a full FE model of the HI-STORM UMAX ISFSI as shown in Figure 3.4.10 is used. The “degraded” elastic modulus of the subgrade under the SFP is appropriately computed to account for the long-term settlement effects as described in the foregoing. Table 3.3.4 lists the bounding subgrade characteristics and the concomitant elastic moduli effective under dynamic loading.

To address different loading patterns on the ISFSI and for completeness, additional partially loaded ISFSI configurations are considered in the evaluations. The partial loaded configurations include a two row loaded ISFSI, a single row loaded ISFSI (the middle row of VVM locations is loaded) and a single VVM loaded ISFSI (a single VVM location centered near the periphery of the ISFSI is loaded). Figures 3.4.14, 3.4.16 and 3.4.18 illustrate the partial loading configurations for the ISFSI. These loading configurations are hereinafter referred to as Simulation Models II, III, and IV, respectively. The effects of exposed Self-hardening Engineered Subgrade between the ISFSI pad and SFP are evaluated in a fifth Simulation Model (Simulation Model V), which is shown in Figure 3.4.11. In this model, the lateral subgrade is completely removed from one side to bound any future excavation activities associated with the construction of a new underground ISFSI. For Simulation Model V, all VVM locations are assumed to be loaded and thus the transporter load is excluded.

To simulate the material continuity at the extreme boundary surface of the substrate under the SFP, translations are constrained at the lateral face of the subgrade. The extreme bottom surface of the model is fixed representing the bedrock (or competent soil) elevation.

The following individual load steps are considered in the analysis:

1. Bounding peak load transmitted by the VVM as determined from the LS-DYNA SSI analysis is applied as an effective pressure on the footprint of the CEC base at all VVM locations.

2. The load from the transporter is applied as a normal pressure (see Figures 3.4.12 and 3.4.16) over the transporter load patch on the ISFSI pad. The transporter is assumed to be positioned over the central VVM cavity.

3. The dead weight from 11’×11’ square ISFSI pad region that is centered above each VVM location (see Licensing Drawing in Section 1.5) is applied as a normal pressure on the SES elements directly beneath the ISFSI pad.

4. To simulate the self-weight of the modeled portion of the ISFSI pad, a 1g gravity load is applied. The densities of the various constituents are appropriately input in the model to accurately reflect the individual component weights.

It must be noted that the structural analysis of the ISFSI conservatively considers the peak dynamic loads from the LS-DYNA SSI analysis. However, it is permissible to use equivalent static loads obtained by removing high frequency components that would not contribute to the structural response using appropriate filters.

Since the peak loads from the LS-DYNA SSI analyses are substantially larger in comparison to the dead and live loads, the load combination LC-3 from Table 2.4.3 governs for the ISFSI structural evaluation. However, the analyses are carried out for load combinations LC-2 and LC-3, and the corresponding results substantiate that the load combination LC-3 is governing.

Figures 3.4.13a through 3.4.21c depict the maximum in-plane stresses in the ISFSI concrete structures for the governing load combination LC-3 for all the ISFSI configurations analyzed. The in-plane axial and bending stress on the SFP and the ISFSI pad elements are post-processed to compute the equivalent moments. The induced moments are compared to the respective moment capacities to determine the corresponding factor of safety. Table 3.4.5 summarizes the results for the SFP and the ISFSI pad respectively for all ISFSI configurations analyzed. The safety factors listed in Table 3.4.5 show that the ISFSI pad contains a substantially greater strength reserve than that required by the ACI code. In establishing the safety factors, no credit has been taken the Dynamic Increase Factor of 25% for flexure and 10% for shear permitted by [3.4.8] in the strength qualification of reinforced concrete under the impactive loadings that are intrinsic to the seismic event.

Table 3.4.6 summarizes the punching shear safety factor for the SFP and the ISFSI pad. The minimum punching shear safety factor under the governing condition (when the loaded transporter is positioned on the ISFSI pad) is found to be well above 1.0.

The peak transporter load on the ISFSI pad from the LS-DYNA SSI analyses plus the load from the ISFSI pad are used to compute the maximum bearing stress in the substrate surface under the ISFSI pad. According to ACI-360 [3.4.9], the bearing stress can be calculated by uniformly distributing the load over the entire bearing area of the pad. The maximum computed bearing stress in the

subgrade below the ISFSI pad (Table 3.4.7) meets the minimum safety factor of 2.0 (suggested by the ACI code [3.4.9]) with ample margin.

Detailed calculations for qualifying the HI-STORM UMAX ISFSI structures are documented in [3.4.17].

Structural Qualification Under the Most Severe Earthquake (MSE)

As described in Subsection 1.0.2, a structurally reinforced version of HI-STORM UMAX has been developed to withstand a substantially stronger earthquake than that analyzed in the foregoing and permitted by the original CoC. The structural reinforcement necessary to maintain robust margins under the stronger earthquake are quite modest as detailed in the licensing drawing package (Section 1.5) wherein the buttressed VVM/ISFSI structure is identified as "Version MSE". The governing earthquake for Version MSE for the long term storage condition is defined by Table 2.3.10. The information presented below seeks to demonstrate that Version MSE can withstand the new DBE event (referred to as the Most Severe Earthquake or MSE) meeting all of the acceptance criteria set forth in Table 2.4.2.

The Design Basis Seismic Model (DBSM) described in the foregoing is utilized to perform the required safety evaluations under the MSE except that, because the earthquake is applied at the base of the Support Foundation Pad (SFP), the SSI model does not include the under-grade (Spaces C and D shown in Figure 2.4.4).

In order to perform the SSI analysis for HI-STORM UMAX Version MSE, five earthquakes, each consisting of statistically independent set of three acceleration time histories (see Table 2.4.5 and Figures 2.4.7 to 2.4.11) are generated. The details on the compliance of the generated earthquakes with the guidelines of SRP 3.7.1 (NUREG-0800) are archived in [2.4.14]. Similar to the SSI analysis performed previously for the DBE condition, the resultant horizontal acceleration time history is used in the SSI analysis for the MSE condition since the DBSM is a symmetric half model. The SSI analysis and subsequent evaluations performed to establish the structural adequacy and safety of Version MSE are summarized below:

i. Strain Compatible Soil Properties: To characterize the structural properties of the subgrade that is suitably conservative for general certification, the properties of the soil from the grade to the Support Foundation Pad elevation at the San Onofre' Nuclear Generating Station (SONGS) are used. The numerical data is provided in the unnumbered table below.

Input Soil Data of the SHAKE Analysis performed for the Most Severe Earthquake						
Soil Layer (numbered from the top surface)	1	2	3	4	5	6
Soil Layer Thickness (ft)	4.5	4.5	4.5	4.5	3	4
Density (lb/ft^3)	130	130	130	130	130	130
Shear Wave Velocity (ft/s)	1000	1100	1200	1300	1440	1500

The 1-D seismic response program SHAKE is used to establish the strain compatible dynamic soil properties for the HI-STORM UMAX underground fuel storage site at the San Onofre' Nuclear Generating Station (SONGS) under the MSE condition. The results obtained for SONGS are adopted as the generic bounding value of the strain compatible soil shear modulus corresponding to the MSE. Since SHAKE solutions for strong earthquake conditions (ZPA>1.0g's) are questionable, a conservative approach is used to determine the dynamic soil properties under the MSE condition for the soil layers above the SFP. First, four discrete SHAKE analyses are performed for Reg. Guide 1.60 earthquakes whose horizontal ZPAs are 0.25g's, 0.5g's, 0.75g's and 1.0g's. Figure 3.4.23 shows the relationship between the average dynamic shear strain of the soil and the earthquake ZPA which is obtained from the four SHAKE analyses. With a resultant horizontal ZPA of 2.121g's, the average soil shear strain under the MSE condition is predicted to be 0.809 in/in based on the extrapolated curve in Figure 3.4.23. Next, the dynamic shear modulus of the soil under the MSE condition is obtained from the shear modulus-shear strain relationship used by SHAKE and shown in Figure 3.4.24. At the predicted shear strain of 0.809 in/in, the modulus reduction factor is approximately 0.075. The corresponding shear wave velocity reduction factor is therefore approximately 0.274 (square root of 0.075). Based on the measured average static shear wave velocity of 1257ft/s at the SONGS, this reduction factor leads to the "best estimated" strain compatible dynamic shear wave velocity of 344 ft/s (1257ft/s × 0.274 = 344ft/s). The uncertainty of the soil shear modulus is considered in the SSI analysis by multiplying the best estimate value by a factor of 1.5 and 1/1.5 to establish the upper bound value and lower bound value, respectively, per ASCE 4-98.

ii. Sensitivity Study on the Lateral Extent of the SSI Model: It is recalled that the SSI model of the ISFSI structure (Space B) has a circular boundary with 700 feet radius. Recognizing that certain sites, such as SONGS, don't have such a large land mass around the ISFSI, a sensitivity analysis on the size of the ISFSI SSI model was performed using the time history set identified as THS #1 in Table 2.4.5. Figure 3.4.25 and Figure 3.4.26 show the "large" Space B SSI model (i.e., Space B size is identical to that used in the initial qualification) and the "small" Space B SSI model (i.e., the Space B lateral boundary is set at 100 feet away from the ISFSI pad), respectively. The key seismic response parameters obtained from the

two models using an identical boundary condition (transmitting boundary) are summarized in Table 3.4.10 wherein the case identifier contains the letter L or S to indicate the "large" and "small" model, respectively. These SSI results (i.e., the results for Cases 1L and 1S) demonstrate that the large model yields a uniformly greater response. Therefore, it is concluded that, regardless of the size of the ISFSI land area at a site, it is conservative to use the "large" model. The analyses summarized below accordingly utilize the "large" ISFSI SSI model.

iii. Governing MPC Storage Condition: A perusal of the results presented in the foregoing for the SSI model under the DBE event listed in Table 2.3.2 showed that the storage scenario wherein all storage cavity locations are occupied by loaded MPC-37s provides the largest response. Therefore, the same loading state is used in the qualification analyses for the MSE event.

iv. Effect of Concrete Cracking: The effect of the SFP concrete cracking was evaluated previously for the DBE condition following the guidelines from ASCE 4-98 [3.4.14] and ASCE/SEI43-05 [3.4.15]. Table 3.4.2 lists both the cracked scenarios (i.e., Scenarios 2 and 4) and the corresponding uncracked scenarios (i.e., Scenarios 1 and 3) considered in the LS-DYNA SSI analysis for the DBE event. The DBE analysis results summarized in Table 3.4.3 consistently demonstrate that the key seismic response (i.e., the seismic impact loading on the SFP) under the uncracked condition is bounding. Therefore, the LS-DYNA SSI analysis performed for the MSE event only considers the governing uncracked concrete condition.

v. SSI Analysis Base Runs: The Design Basis Seismic Model used in the original certification and the governing MPC storage condition along with the established "best estimate" soil properties identified above are used to analyze the response of the representative "UMAX Version MSE" array under the five earthquakes composed of statistically independent time histories shown in Figures 2.4.7 to 2.4.11(i.e., Cases 1L, 2L, 3L, 4L and 5L in Table 3.4.10). These five discrete seismic analyses using LS-DYNA are referred to as "base runs".

vi. Uncertainty in the Subgrade (Space B) Properties: The uncertainty of the soil properties is studied through two additional SSI simulations for the 2nd earthquake, which has the largest horizontal ZPA per Table 2.4.5, by using the calculated lower bound and upper bound modulus values. The object of the parametric runs is to determine the extent that the key response parameters are affected by the change in the subgrade modulus. Specifically, the ratio of each response parameter for the upper and lower bound soil modulus to its base run value is obtained. The largest value of the ratio for each response parameter is referred to as its amplification factor. If a particular response parameter is governed by the best estimate soil properties (i.e., response is less for upper and lower bound properties), the amplification factor is set equal to 1.0. The two additional SSI simulations, i.e., Cases 2LL and 2LU in Table 3.4.10 with the last letter in the case number indicating L (for lower bound soil property) and U (for upper bound), provide the needed information to ascertain the amplification factor (due to the uncertainty in the subgrade property) for each response

parameter. These amplification factors are applied to the results from the base run to establish the bounding response parameter for the structural qualification analysis of the HI-STORM UMAX Version MSE array. In this manner the effect of uncertainty in the subgrade modulus is incorporated in the safety evaluation.

vii. Determination of Governing Seismic Response Parameters: The amplification factor determined based on the soil property variation for each seismic response parameter is listed in Table 3.4.11. The averaged peak seismic response parameters obtained from the SSI analyses using the "best estimate" soil properties for the five earthquakes (i.e., the averaged values of Cases 1L through 5L) are also listed in that table. The governing seismic response parameters, which provide the input data for the HI-STORM UMAX Version MSE structural qualification analysis, are then determined by multiplying the averaged values with the corresponding amplification factors.

viii. Structural Evaluations: Structural analyses are performed for HI-STORM UMAX Version MSE by using the same analysis methods as those employed in the initial HI-STORM UMAX qualification. Detailed calculations are documented in [3.4.1] and [3.4.17]. As before, the safety analysis focuses on the following parts deemed to be most severely stressed in the system, namely, (a) The Top and Bottom MPC Guide plates and Guide Ring, (b) The MPC Shell (primary stress from beam-bending action), (c) Local stress in the MPC Shell from impact with the Guide Plate, (d) The uplift load in the Closure lid hold-down structure and (e) The flexural stress in the Support Foundation Pad from the amplified vertical inertia load during the MSE event. The summary evaluations, synopsized from the respective Calculation Packages, are presented below:

a. Top & Bottom MPC Guide Plates and Guide Ring: As shown in the licensing drawing package (Section 1.5), the Top and Bottom Guide Plates Guide Ring are configured to close their gap with the MPC surface and thus eliminate impact loads from rattling action of the MPC during the earthquake. Despite this measure, the maximum MPC horizontal acceleration is in excess of 8 G's and the maximum compressive load on the Guide Plate (the top Guide Plate load controls) is approximately 470 kips, as reported in Table 3.4.11. The computed capacity of the Guide Plates under the worst direction of compressive loading, as reported in Table 3.4.12, is greater than 530 kips resulting in a factor of safety in excess of 1.1.
b. The maximum MPC shell primary stress computed from the LS-DYNA analyses, reported in Table 3.4.11, likewise shows a positive margin of safety even without accounting for the strength increase of the stainless steel material under dynamic conditions.
c. The local strain in the impacted region of the MPC shell, reported in Table 3.4.12, is also only a fraction of the limit set forth in Table 3.1.1.
d. The hold-down structure, illustrated in the licensing drawing in Section 1.5, is computed to have a large factor of safety (Table 3.4.12).
e. The factor of safety in the Support Foundation Pad, computed by simply proportioning the maximum vertical load for the previously analyzed DBE event and

the MSE event, as shown in Table 3.4.13, is likewise large for reinforced concrete structures. The values shown in Table 3.4.13 are valid since the analysis approach used to determine the seismic demand loads for the MSE condition is conservative as compared to the original analysis for the DBE condition. Specifically, in the DBE analysis the earthquake motion is deconvoluted from the free field surface down to the bedrock level (100 ft below grade) using the computer code SHAKE2000. The deconvoluted earthquake motion obtained from SHAKE2000 is then applied as the control motion at the bedrock level in the LS-DYNA soil-structure interaction (SSI) model. In the MSE analysis, the free field acceleration is applied directly to the SFP level (25 ft below grade) without considering any decrease in earthquake strength due to deconvolution. Recognizing that the calculated seismic demand loads for the MSE condition are conservative relative to the original results for the DBE condition, and that the ANSYS model used to evaluate the reinforced concrete ISFSI structure is linear, the minimum factors of safety for the SFP for the MSE condition are obtained by linearly scaling the previous results for the DBE condition according to the seismic demand ratio.

f. The ovalization of the Cavity Enclosure Container CEC) shell is negligible.

Safety Conclusions:

The above analyses show that the Version MSE of HI-STORM UMAX meets all applicable structural acceptance criteria. In particular, it is shown that:

a. The MPC retrievability subsequent to the MSE event is maintained with a large margin.
b. The maximum primary stress in the MPC shell remains well below the Level D primary stress intensity limit for Class 1 “NB” pressure vessels.
c. The local impact strain in the MPC enclosure vessel is a small fraction of the local strain limit for stainless steel specified in Table 3.1.1.
d. The bottom and top MPC guide plates remain functional during the MSE condition.
e. The maximum MPC deceleration is less than the acceptable value listed in Table 3.1.1.
f. The structural strength of the ISFSI slab (Table 2.4.3) is not exceeded.
g. The puncture strength of the ISFSI slab is not exceeded.
h. The structural capacity of the VVM lid to CEC top flange connection is not exceeded which ensures that the Closure Lid will not, even temporarily, lift off the ISFSI pad.

3.4.4.1.3 Design Basis Missile Loading (Load case 02 in Table 2.4.1)

3.4.4.1.3.1 Tornado Missile Strike on VVM Closure Lid (Load Case 02 in Table 2.4.1)

Design basis tornado missiles are specified in Table 2.3.3. The Closure Lid is the only component of the VVM that is accessible to a tornado missile; therefore, missile impact analyses focus on this component. The impact a large missile is evaluated to determine whether the Closure Lid can maintain its required shielding function. Because of its size and the steel cross structure in the central pipe of the Closure Lid, the large missile can only deliver its kinetic energy to the side and

top surface of the lid and with no penetration. The medium missile, however, is able to pass through the top opening in the Closure Lid, impacting what is termed the "Lid Extension", a concrete filled steel weldment.

The formula from "Topical Report – Design of Structures for Missile Impact", BC-TOP-9A, Rev. 2, 9/74 [3.4.12]] is used to establish appropriate pressure-time data. For the speed and mass associated with the large missile, the impact force-time curve has the form

$$F(t) = 0.625 \text{ sec/ft} \times 184.8 \text{ ft/sec} \times 4000 \text{ lb} \times \sin(20t) = 462{,}000 \text{ lb} \times \sin(20t) \text{ for } t < 0.0785 \text{ sec.}$$
$$= 0 \text{ for } t >= 0.0785 \text{ sec.}$$

This representation of the large missile impact load is appropriate as demonstrated by the result of a modern passenger vehicle full-scale impact testing. Figure 3.4.22 shows the force-time history from the full-scale test of a full-size Ford passenger vehicle [3.4.11]. The test was performed at an impact speed of 35 mph and the vehicle had approximately the same weight as the design basis large deformable missile. Since the force is directly proportional to the pre-impact momentum, an estimate of the peak force at 126 mph for the vehicle is obtained by a simple ratio of the impact velocities and missile mass. Estimating the peak value from the plot produces a resulting peak force is found to be in good accord with the peak value predicted from [3.4.12], which provides a good confirmation of the methodology in [3.4.12]. Because of the simple geometric configuration of the Closure Lid, the large missile impact is evaluated using strength of materials approach. The calculation results indicate that the large missile event will neither cause the collapse of the Closure Lid nor dislodge the lid from the CEC cavity. Moreover, the lid concrete bearing capacity will not be exceeded by the large missile impact. The details of this calculation are found in [3.4.1].

The analyses performed for the intermediate missile (i.e., a rigid 275 lb 8" diameter cylindrical steel bar) and the small missile (i.e., a 1" diameter solid steel sphere) are based on the energy approach previously used in the missile impact analyses for HI-STORM 100 and HI-STORM FW storage systems. No credit for the central steel "cross" shaped structure is taken and the impact on the "Lid Extension" is also considered in the analysis. Analysis results demonstrate that the intermediate and small missiles will not penetrate the steel weldment and encased concrete of the Closed Lid or cause the drop of the Extended Lid to threat the stored MPC.

A summary of results are presented in Table 3.4.8. Thus, the assessment of all missile impact scenarios leads to the conclusion that the postulated missile strikes will not preclude MPC retrievability, will not cause loss of confinement, and will not affect criticality.

3.4.4.1.3.2 Tornado Missile Protection During Construction

The scenario of an excavation near the ISFSI has been considered. The optional Enclosure Wall on the exposed side of the ISFSI is not considered in the tornado missile analysis. The impact of the exposed lean concrete by the Design Basis Missiles (Table 2.3.3) using the methodology documented in [3.4.10] shows that the MPC storage cavities remain beyond the reach of the missiles.

3.4.4.1.4 Non-Mechanistic Tipover

Tipover is not an applicable load case for HI-STORM UMAX. The VVM is situated underground and cannot be moved; therefore, drop and tipover events are not credible accidents for this design configuration.

3.4.4.1.5 Maximum Temperature and Internal Pressure under Normal and Off-Normal Conditions

The HI-STORM UMAX VVM is open to the environment; therefore, it is not subject to any internal pressure.

The MPCs authorized for storage in HI-STORM UMAX (listed in Chapter 1) have been analyzed and qualified for normal and off-normal conditions of storage in their host docket (i.e., docket number 72-1032 for HI-STORM FW). Calculations in Chapter 4 show that the internal operating temperature in every MPC at its permissible maximum heat load when stored in HI-STORM UMAX is less than the value used in the stress analysis in docket number 72-1032. Because the internal pressure in the MPC bears a proportional relationship to the average internal temperature in the MPC cavity, it follows that the internal pressure in the MPC when stored in HI-STORM UMAX will be less than that assumed in the stress analysis of the MPC in docket number 72-1032. Therefore, the stress values computed for the MPC under normal and off normal operating conditions in docket number 72-1032 will envelope their corresponding values for storage in HI-STORM UMAX.

3.4.4.1.6 Maximum Temperature and Internal Pressure Under Accident Conditions

HI-STORM UMAX, being an open to environment cask, does not experience any internal pressure under accident conditions.

The MPCs authorized for storage in HI-STORM UMAX (listed in Chapter 1) have been analyzed and qualified for accident conditions of storage in their host docket (i.e., docket number 72-1032 for HI-STORM FW). Calculations in Chapter 4 show that the internal operating temperature in every MPC at its permissible maximum heat load when stored in HI-STORM UMAX is less than the value used in the stress analysis in docket number 72-1032. Because the internal pressure in the MPC bears a proportional relationship to the average internal temperature in the MPC cavity, it follows that the internal pressure in the MPC when stored in HI-STORM UMAX will be less than that assumed in the stress analysis of the MPC in docket number 72-1032. Therefore, the stress values computed for the MPC under accident conditions in docket number 72-1032 will envelope their corresponding values for storage in HI-STORM UMAX.

3.4.4.1.7 Handling of Components

The stress analyses of the HI-STORM UMAX VVM under normal handling conditions are presented in Subsection 3.4.3. The stress analyses of the MPC and the HI-TRAC transfer cask under

normal handling conditions are presented in their host docket (i.e., docket number 72-1032 for HI-STORM FW).

3.4.4.1.8 Snow Load

Load Case 01 in Table 2.4.1, presented later in this section, is a bounding load combination that conservatively subsumes a number of normal and extreme environmental phenomena loads including snow load.

3.4.4.1.9 Transfer Cask and Mating Device Loading on VVM Container Shell (Load Case 06 in Table 2.4.1)

During HI-STORM UMAX system loading, a HI-TRAC transfer cask with a fully loaded MPC is placed over a HI-STORM UMAX VVM using a specially designed transporter and a lifting device meeting ANSI N14.6 stress margin requirements. The transfer cask is connected to the CEC using an ancillary mating device (see Chapter 9). Although the Self-hardening Engineered Subgrade will not settle relative to the SFP during the entire service time of the ISFSI, the CEC shell is evaluated herein by conservatively assuming that the entire weight of the transfer cask and the mating device is supported by the shell and that neither the ISFSI pad nor the SES provides any support to the CEC shell. Based on a conservatively assumed total weight (400 kips) of the transfer cask and mating device, the compressive stress of the CEC shell is compared with the Level A service stress limit for NF Class 3 plate and the critical buckling stress for thin-walled cylindrical shell. Results presented in Table 3.4.9 demonstrate that the CEC shell can support the total weight of transfer cask and mating device with large margins.

3.4.4.1.10 Dead Load plus Design Basis Explosion Pressure on VVM Components (Load Case 01 in Table 2.4.1)

The VVM Closure Lid rests on the CEC and resists vertical loads, arising from dead weight, and from induced loadings from explosions, from seismic accelerations, and from tornado missile impact. In this subsection, the analysis considers only the normal loading condition plus a steady pressure that bounds the explosion pressure (see Table 2.3.1). Due to the simple configuration of the lid, the strength of materials method is used perform the evaluation. The stresses from the solution are compared, per the criteria in Table 2.4.1, with allowable stress values for plate and shell structures as provided in ASME Section III Code, Subsection NF. The allowable stress intensity is per Table 3.1.6 for Level D conditions.

As demonstrated in the structural calculation package [3.4.], the combined load from dead weight and the design basis explosion pressure is bounded by the maximum missile vertical impact force applied to the lid. Moreover, it's impossible for the CEC shell to deform under the design basis explosion pressure since it is directly against the SES with a bearing capacity of at least 10 times the design basis explosion pressure. It is therefore concluded that there is a large margin of safety in the HI-STORM UMAX Closure Lid and CEC Container Shell against the lateral pressure from Design Basis Explosion.

3.4.4.1.11 Design Basis Fire on VVM Closure Lid (Load Case 05 in Table 2.4.1)

With respect to the fire event (Load Case 05 in Table 2.4.1), where the Closure Lid steel temperature is conservatively assumed to rise to the limit set in Table 2.4.1, it is noted from Tables 3.1.4 and 3.3.1 that the Level A stress limit is reduced to 0.60 of the room temperature value, the yield strength is reduced to 0.67 of its room temperature value, and the ultimate strength is reduced to 0.92 of its room temperature value. From the stress values obtained in the lid (even with the explosion 10 psi surface pressure load included), it is evident that a structural collapse of the lid due to reduction of the ultimate strength because of the heat of the fire is not possible.

3.4.5 Cold

Due to its subterranean configuration, the structural components of the VVM are relatively protected from extremes in the ambient temperature in comparison to the above ground HI-STORM certified in docket number 72-1032. Therefore, no new analyses are identified for the HI-STORM UMAX system.

3.4.6 Miscellaneous Evaluations

None.

3.4.7 Service Life of HI-STORM UMAX VVM

The term of the 10CFR72, Subpart L C of C, granted by the NRC is 20 years; therefore, the License Life (see Glossary) of all components is 20 years. The principal design considerations that bear on the adequacy of the storage module for the service life are addressed as follows:

Exposure to Environmental Effects

All exposed surfaces of the HI-STORM UMAX Cavity Enclosure Canister that are made from ferritic steels are painted and protected from corrosion on the outside by appropriate means as described in Chapter 8. The inside surface of the CECs is protected by paint or the CEC is made of stainless steel. The Divider Shells are protected by paint or are made of stainless steel. In addition, one side of the Divider Shell is further protected by insulation. Concrete, which serves strictly as a shielding material in the Closure lid, is completely encased in steel. Under normal storage conditions, the bulk temperature of the HI-STORM UMAX system will, because of its large thermal inertia, change very gradually with time. Therefore, material degradation from rapid thermal ramping conditions is not credible for the HI-STORM UMAX system. The configuration of the storage modules assures resistance to freeze-thaw degradation. In addition, the storage modules are specifically designed for a full range of enveloping design basis natural phenomena that could occur over the 60-year design life of the storage system. Chapter 8 provides further discussion on chemical and galvanic reactions, material compatibility and operating environments pertaining to HI-STORM UMAX.

Material Degradation

As discussed in Chapter 8, the relatively low neutron flux to which the storage modules are subjected is insufficient to produce measurable degradation of the cask's material properties and impair its intended safety function. Exposed carbon steel components are coated to prevent corrosion. The controlled environment of the ISFSI storage pad mitigates damage due to direct exposure to corrosive chemicals that may be present in other industrial applications.

Maintenance and Inspection Provisions

The requirements for periodic inspection and maintenance of the storage VVM throughout the 60-year design life are defined in Chapter 10. These requirements include provisions for routine inspection of the inlet plenums and periodic visual or camera aided verification that the air flow paths are free and clear of debris. ISFSIs located in areas subject to atmospheric conditions that are particularly aggressive should be evaluated by the licensee on a site-specific basis to determine the frequency for such inspections to assure long-term performance. In addition, the HI-STORM UMAX system is designed for easy retrieval of the MPC from the storage VVM should it become necessary for any reason.

In summary, the VVM is engineered for a 60 year design life and a 100 year service life, while satisfying the conservative design requirements defined in Chapter 2. For information supporting the 60 year design life addressing chemical and galvanic reactions as well as other potentially degrading factors, reference is made to Chapter 8. Requirements for periodic inspection and maintenance of the HI-STORM UMAX VVM throughout the 60-year design life are defined in Chapter 10. The VVM is designed, fabricated, and inspected under the comprehensive Quality Assurance Program discussed in Chapter 14 and docket number 71-0784 (The service life of a system may exceed its design life if the system is maintained in accordance with the supplier's O&M manual).

The above findings supporting the HI-STORM UMAX service life are consistent with those of the NRC's Waste Confidence Decision Review [3.4.13], which concluded that dry storage systems designed, fabricated, inspected, and operated in accordance with such requirements are adequate for a 100-year service life while satisfying the requirements of 10CFR72.

Table 3.4.1

KEY STRESS RESULTS FOR HI-STORM UMAX CLOSURE LID
NORMAL HANDLING

Item	Calculated Value (ksi)	Allowable Limit (ksi)	Safety Factor
Lifting Hole Tearout Stress	1.464	1.52	1.04
Lifting Hole Bearing Stress	3.833	4.56	1.19
Governing Weld Joint Stress	2.530	3.04	1.20

Table 3.4.2

SEISMIC LOADING SCENARIOS ANALYZED FOR HI-STORM UMAX

Scenario 1	All storage locations loaded with maximum weight MPCs, and a loaded VCT is placed at the center of the ISFSI.
Scenario 2	Same as Scenario 1 except that the Young's Modulus of the SFP concrete is reduced to one-half of its nominal value.
Scenario 3	Same as Scenario 1 except that the subgrade adjacent to one side of SES (Space A) is excavated down to the SFP and that the VCT is not considered.
Scenario 4	Same as Scenario 3 except that the Young's Modulus of the SFP concrete is reduced to one-half of its nominal value.

Table 3.4.3

ISFSI INTERFACE LOADS OBTAINED FROM LS-DYNA SSI SIMULATIONS

Interface Load	Scenario 1	Scenario 2	Scenario 3	Scenario 4
CEC to SFP Impact Load[1], lbf	7.7191×10^5	7.1945×10^5	7.1505×10^5	6.6774×10^5
Transporter to ISFSI Pad Contact Load per Track[2], lbf	8.6014×10^5	8.0838×10^5	NA	NA

Notes:
1. A bounding value of 8.0×10^5 lbf is conservatively used in the strength qualification of ISFSI structure reported in subparagraph 3.4.4.1.2;
2. A bounding value of 1.0×10^6 lbf is conservatively used in the strength qualification of ISFSI structure reported in subparagraph 3.4.4.1.2.

Table 3.4.4

HI-STORM UMAX VVM COMPONENTS SEISMIC QUALIFICATION ANALYSIS RESULTS

Item	Calculated Value	Allowable Limit	Safety Factor
Ovalization of VVM Shells	Not a credible concern; see discussion in subparagraph 3.4.4.1.2		
MPC Shell Primary Stress, ksi	25.88	42.0	1.62
MPC Guide Impact Load, lbf	2.1161×10^{5}†	2.368×10^{5}	1.12
MPC Peak Impact Deceleration, g's	27.854	Table 3.1.1	2.2.2
Local Plastic Strain of MPC Shell, in/in	0.028	0.1	3.57

† This is the maximum impact load obtained from the four SSI runs listed in Table 3.4.2 and is bounded by the corresponding impact force (see Figure 3.4.6a) in the governing MPC to MPC guide impact analysis.

Table 3.4.5

MOMENT RESULTS AND CORRESPONDING MINIMUM SAFETY FACTORS FOR THE ISFSI STRUCTURES

ISFSI Load Configuration	Moment Induced (lbf-in/in)	Axial Force (lbf/in)	Corresponding Moment Capacity (lbf-in/in)	Minimum Safety Factor
Support Foundation Pad (SFP) ‡				
Simulation Model I	84,841	3095.8	207,792	2.449
Simulation Model II	94,625	-584.9	254,460	2.689
Simulation Model III	110,480	993.9	235,770	2.134
Simulation Model IV	96,271	770.3	238,750	2.479
Simulation Model V	78,835	4,502	189,080	2.398
ISFSI Pad ‡				
Simulation Model I	94,265	-39855	274,688	2.914
Simulation Model II	89,244	-14741	339,910	3.808
Simulation Model III	88,090	-31496	323,170	3.668
Simulation Model IV	87,086	-27938	343,800	3.947
Simulation Model V	62,025	4150	171,970	2.773

‡ The moment capacities for the SFP and ISFSI pad are calculated using axial-force-moment interaction diagram corresponding to the axial force and moment induced in the limiting element. Figure 3.4.13a through 3.4.21c also capture the stress plots for the governing load combination (LC-3 in Table 2.4.3) for all the ISFSI loading configurations analyzed.

Note that the flexural safety factors calculated above are based on the maximum moment induced in a single element, which is very conservative. Averaging over a width of the loaded section would result in much higher safety factors.

Table 3.4.6

MINIMUM SAFETY FACTORS AGAINST PUNCHING SHEAR FAILURE FOR THE ISFSI STRUCTURES

ISFSI Structure	Punching Shear Safety Factor
SFP	2.8
ISFSI Pad	1.5

Table 3.4.7

BEARING STRESS OF SUBGRADE UNDER THE ISFSI PAD

Computed Bearing Stress (psi)	Allowable Bearing Stress (psi)	Safety Factor	Minimum Safety Factor Required per [3.4.9]
38.7	100†	2.58	2.0

† Table 3.3.4 lists the actual bearing stress capacity, which is much greater than this conservatively specified allowable bearing stress.

Table 3.4.8

MISSILE IMPACT ANALYSIS RESULTS

Missile Type – Impact Location	Item	Calculated Value	Allowable Limit	Safety Factor
Large Missile – Impact on HI-STORM UMAX Closure Lid	Horizontal impact load that may dislodge the Closure lid, lbf	4.856×10^5	1.574×10^6	3.24
	Vertical impact load that may lead to the collapse of the Closure lid, lbf	3.237×10^5	1.077×10^6	3.33
Intermediate Missile – Impact on HI-STORM UMAX Closure Lid	Penetration, in	3.346	20	5.98
	Vertical impact load that may lead to the drop of the "extended lid", lbf	3.237×10^5	1.203×10^6	3.71
Small Missile – Impact on HI-STORM UMAX Closure Lid	Penetration, in	0.266	20	75.2
Large Missile – Impact on Exposed SES surface	Penetration, ft	0.177	10	56.5
Intermediate Missile – Impact on Exposed SES surface	Penetration, ft	0.544	10	18.4
Small Missile – Impact on Exposed SES surface	Penetration, ft	0.112	10	89.3

Table 3.4.9

BOUNDING STRESS AND BUCKLING ANALYSIS RESULTS OF THE SHELL DURING MPC TRANSFER OPERATION

Item	Calculated Value ksi	Allowable Limit ksi	Safety Factor
Compressive Stress of CEC Shell	1.698	Compression: 16.7	9.8
		Buckling: 252.4	148.7

Table 3.4.10

KEY RESULTS OF LS-DYNA SSI ANALYSES FOR HI-STORM UMAX VERSION MSE UNDER THE MOST SEVERE EARTHQUAKE CONDITION

Case (1)	Maximum MPC Guide Plate Force (kips)	Maximum SFP Load (kips)	SES Maximum Horizontal Acceleration (g's)	Closure Lid Maximum Accelerations (g's) (2)	MPC Maximum Horizontal Accelerations (g's)
1S	326.62	625.9	5.14	6.93 (H) 1.815 (V)	6.4
1L	374.06	808.0	5.89	7.27 (H) 1.51 (V)	7.30
2L	394.57	1,061.2	6.22	8.18 (H) 2.46 (V)	7.99
3L	349.12	875.08	5.57	6.04 (H) 1.88 (V)	7.16
4L	388.49	942.44	6.01	8.20 (H) 1.59 (V)	8.65
5L	446.76	1,232.8	6.64	8.34 (H) 2.48 (V)	7.99
2LL	426.96	933.58	7.13	9.39 (H) 2.39 (H)	8.00
2LU	475.89	999.73	6.96	7.94 (H) 1.63 (V)	9.02

Notes: (1) The SSI analysis case name consists of a number and one or two letters and follows the following convention: The number shows which of the five earthquakes is considered; the first letter "L" or "S" identifies if the "large" or "small" soil" model is considered."Best estimate" soil properties are used unless a second letter "L" or "U" is used in the case identifier to indicate if the "lower" (L) or "upper" (U) bound soil properties are considered. (2) (H) and (V) respectively denote lid accelerations in horizontal and vertical directions.

Table 3.4.11

GOVERNING SSI RESPONSE PARAMETERS UNDER THE MOST SEVERE EARTHQUAKE CONDITION

Item	Maximum Impact Force on the MPC Guide Plate (kips)	Maximum Compression Load on the Support Foundation Pad (SFP) (kips)	Maximum Horizontal Acceleration in the Concrete in Space A (g's)	Closure Lid Maximum Accelerations (g's)	MPC Maximum Horizontal Accelerations (g's)
Averaged Response of 5 Earthquakes	390.60	984.10	6.07	7.61 (H) 2.07 (V)	7.76
Amplification factor	1.21	1.0	1.15	1.15 (H) 1.0 (V)	1.13
Governing Response Parameters	471.10	984.10	6.95	8.73 (H) 2.07 (V)	8.76

Table 3.4.12

STRUCTURAL QUALIFICATION ANALYSIS RESULTS OF HI-STORM UMAX VVM COMPONENTS UNDER THE MSE CONDITION

Item	Calculated Value	Specified Limit for Safety Evaluation	Safety Factor
Ovalization of VVM Shells	Not a credible concern; due to the upgraded SES material (3,000 psi concrete) for the MSE condition		
MPC Shell Primary Stress, ksi	35.92	42.0	1.17
MPC Guide Impact Load, lbf	4.711×10^{5} (1)	5.352×10^{5} (2)	1.14
MPC Peak Impact Deceleration, g's	44.609 (3)	61.75 (4)	1.38
Local Plastic Strain of MPC Shell at the Impact Location in the Lid Region, in/in	0.025	0.1 (See Table 3.1.1)	4.0
Closure Lid Hold-Down Structure Uplift Load, kips	31.0	99.6(5)	3.21

Notes: (1) This is the maximum impact load obtained from the governing seismic response results listed in Table 3.4.11 and is bounded by the corresponding impact force (see Figure 3.4.27) obtained in the bounding MPC to MPC guide impact LS-DYNA analysis. (2) Minimum value of the allowable limits of the top MPC guide plate and the bottom MPC guide ring. (3) Conservatively taken as the peak MPC deceleration in the bounding MPC to MPC guide impact LS-DYNA analysis. (4) The limit on the MPC peak deceleration utilizes the computed value for the non-mechanistic tip-over event in the HI-STORM FW docket. (5) Based on the capacity of the weakest component in the lid hold-down structure.

Table 3.4.13

MINIMUM SAFETY FACTOR OF THE SUPPORT FOUNDATION PAD UNDER THE MSE CONDITION

Seismic Condition	Peak Seismic Load Applied to the SFP Through A Loaded MPC (kips)	Minimum SFP Safety Factor
Design Basis Earthquake	800.0	2.13 (1)
Most Severe Earthquake	984.1	1.73 (2)

Notes: (1) The minimum SFP safety factor taken from Table 3.4.5; (2) The SFP minimum safety factor under the MSE condition is conservatively obtained by multiplying the minimum safety factor computed previously under the DBE condition (in the initial qualification) by the ratio of the maximum seismic load on the SFP under DBE to the corresponding load computed under MSE.

CERTIFICATE OF SERVICE

I hereby certify that on July 20, 2020, I electronically filed the foregoing with the Clerk of the Court for the United States Court of Appeals for the Ninth Circuit by using the appellate CM/ECF system.

Participants in the case who are registered CM/ECF users will be served by the appellate CM/ECF system.

Dated: July 20, 2020

Respectfully submitted,
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No. 20-70899

IN THE UNITED STATES COURT OF APPEALS
FOR THE NINTH CIRCUIT

IN RE PUBLIC WATCHDOGS,

Petitioner,

v.

UNITED STATES NUCLEAR REGULATORY COMMISSION,

Respondent.

SOUTHERN CALIFORNIA EDISON COMPANY,
Intervenor.

INTERVENOR SOUTHERN CALIFORNIA EDISON'S SUPPLEMENTAL EXCERPTS OF RECORD

VOLUME 6 OF 8

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SUPPLEMENTAL EXCERPTS OF RECORD:

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[PROPRIETARY INFORMATION WITHHELD IN ACCORDANCE WITH 10 CFR 2.390]

Figure 3.4.1; 3-D LSDYNA Soil Model for the Design Basis Seismic Response Analysis

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Figure 3.4.2; 3-D LSDYNA Model for the Non-Linear SSI Analysis of the HI-STORM UMAX ISFSI under the Loading Scenarios 1 and 2 in Table 3.4.2

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Figure 3.4.3; 3-D LSDYNA Model for the Non-Linear SSI Analysis of the HI-STORM UMAX ISFSI under the Loading Scenarios 3 and 4 in Table 3.4.2

[PROPRIETARY INFORMATION WITHHELD IN ACCORDANCE WITH 10 CFR 2.390]

Figure 3.4.4; Maximum Tensile Stress Experienced by the Exposed SES due to Excavation during the DBE Event

[PROPRIETARY INFORMATION WITHHELD IN ACCORDANCE WITH 10 CFR 2.390]

Figure 3.4.5; LSDYNA Model for the Governing MPC to Guide Impact

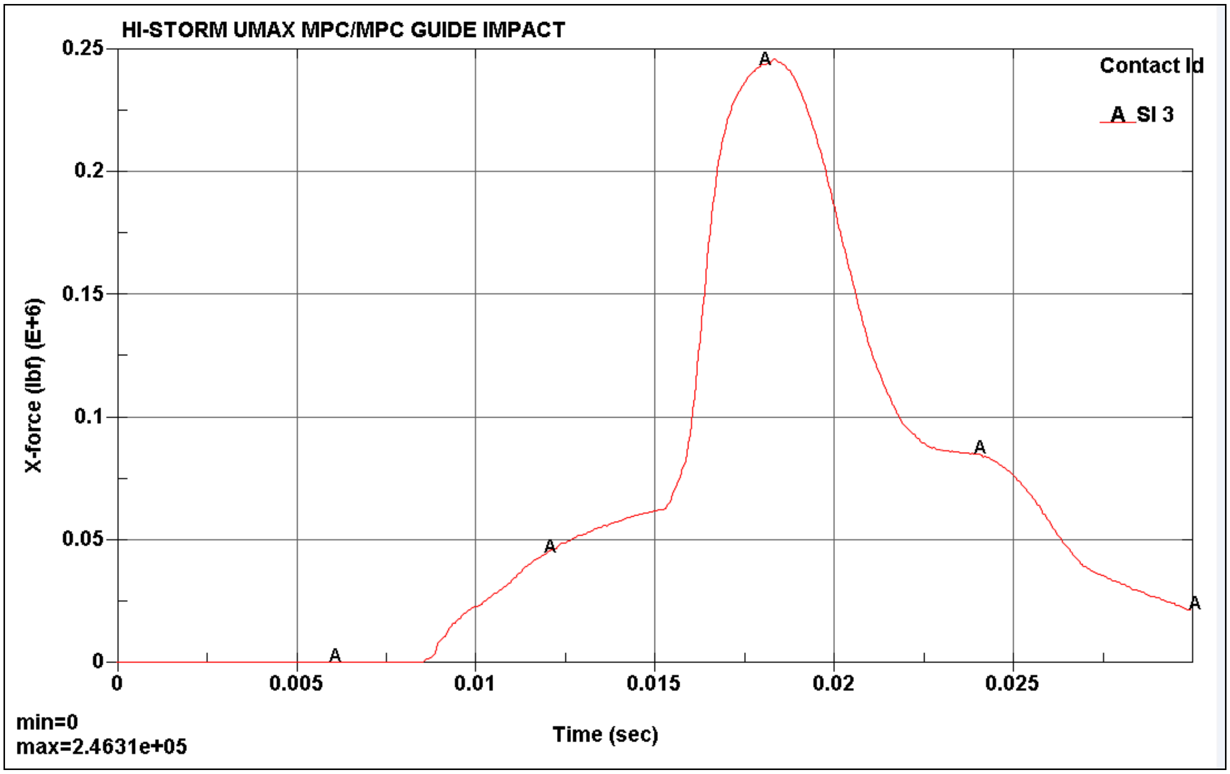


Figure 3.4.6a; Time History of the Impact Force at the MPC/MPC Top Guide Interface
(Maximum force = $2\times2.4631\times10^5$ lbf = 4.9262×10^5 lbf due to the half model)

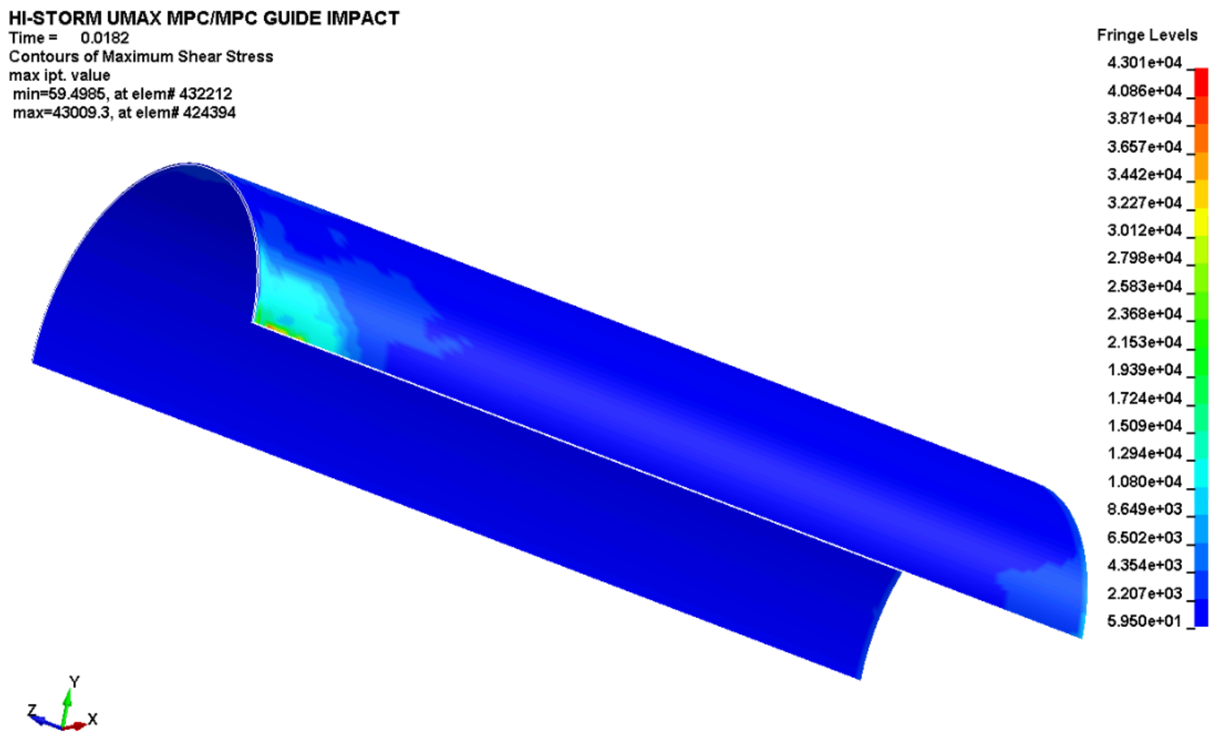


Figure 3.4.6b; Maximum Shear Stress of the MPC Shell
(Maximum Primary Stress Intensity = 2×12,940 psi = 25,880 psi)

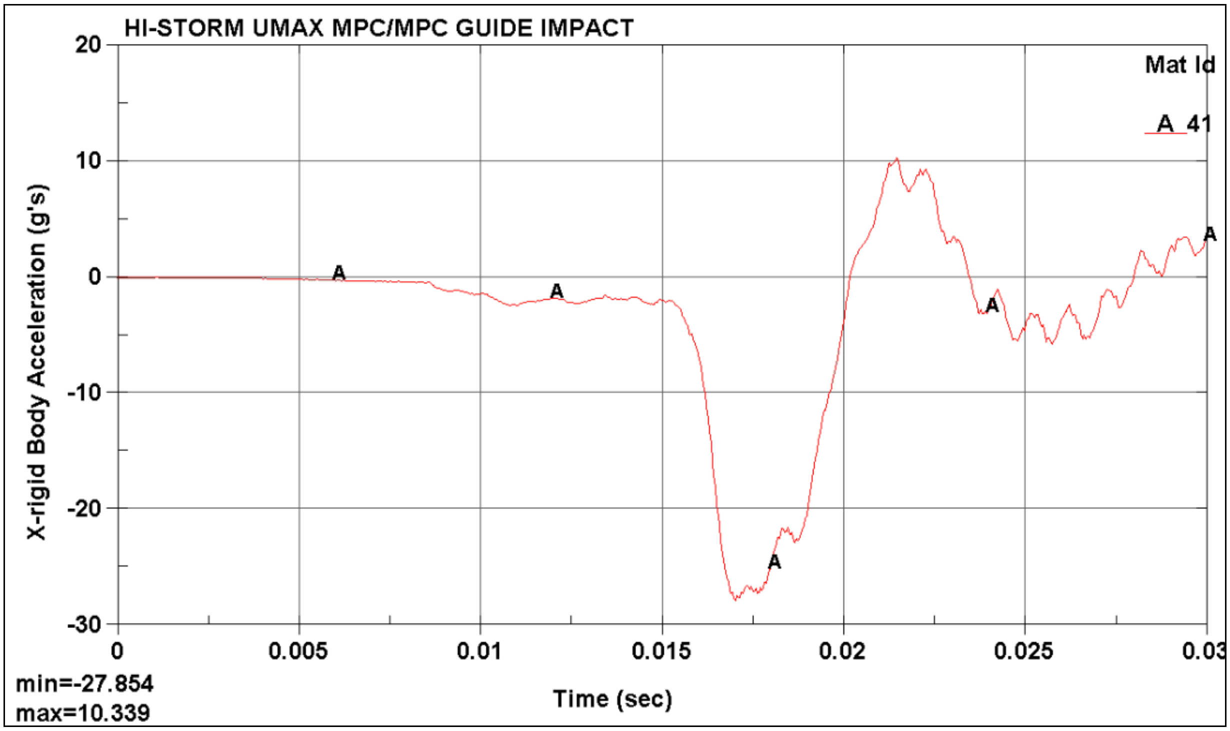


Figure 3.4.7; MPC Top Lid Impact Deceleration Time History under the DBE Condition

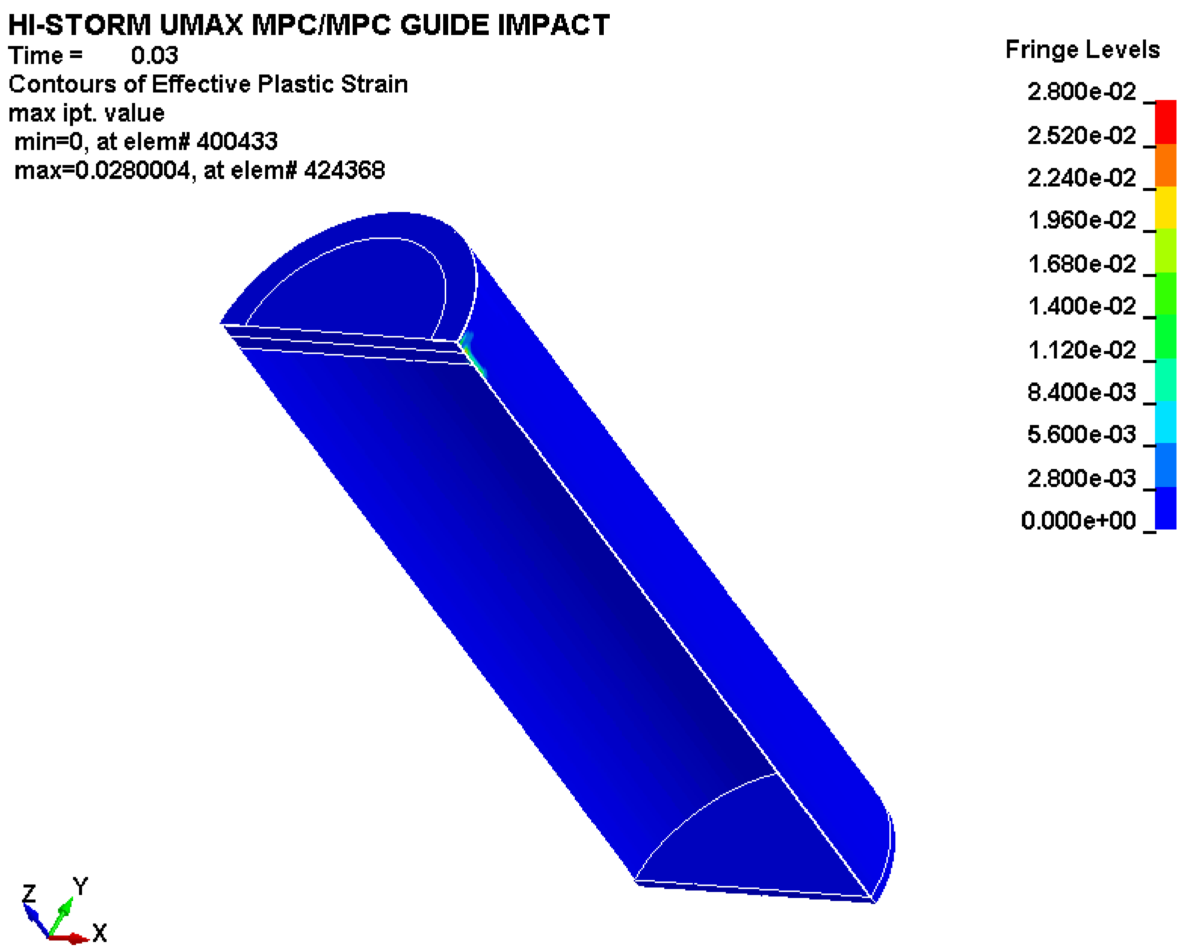


Figure 3.4.8; Maximum Plastic Strain of the MPC Enclosure Vessel Due to the Bounding Impact with the MPC Top Guide under the DBE Condition

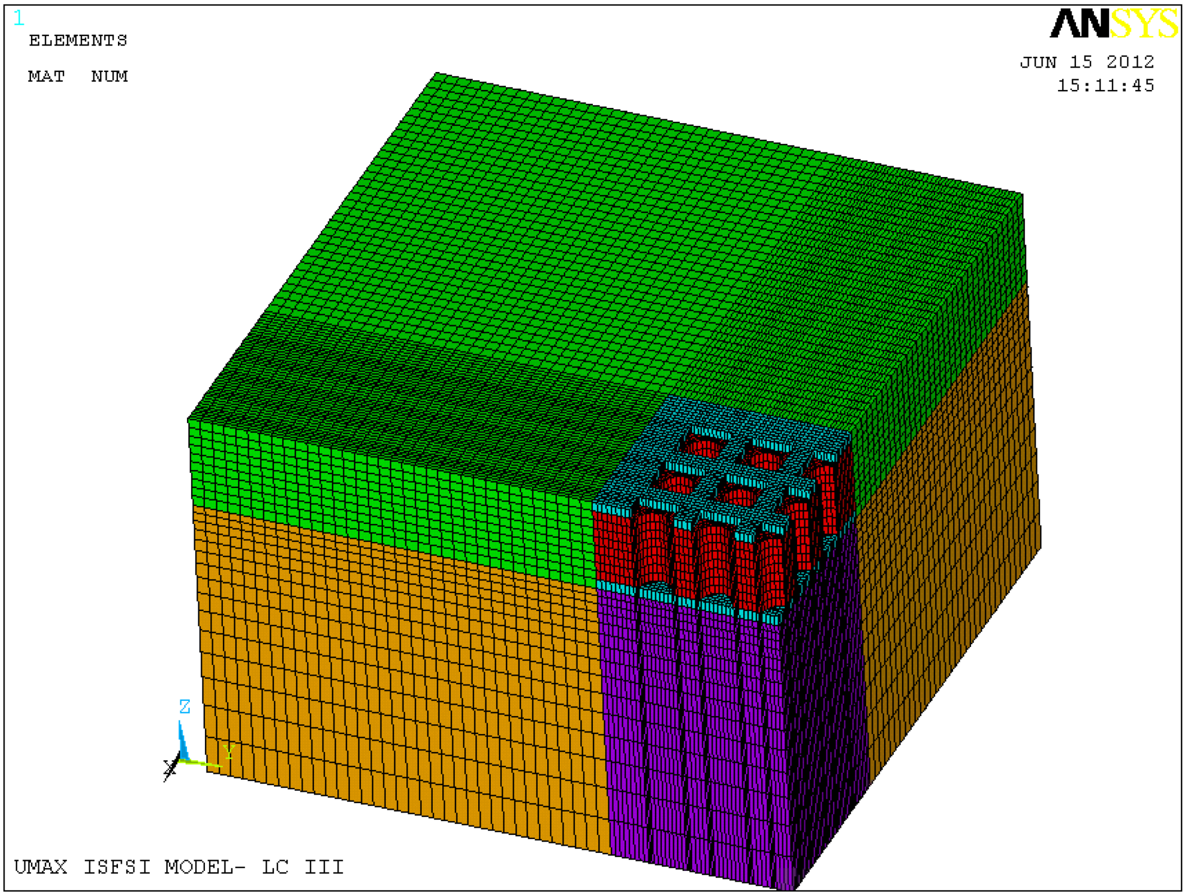


Figure 3.4.9; Finite Element Model of the ISFSI Reinforced Concrete Structures for Simulation Models I, III and IV

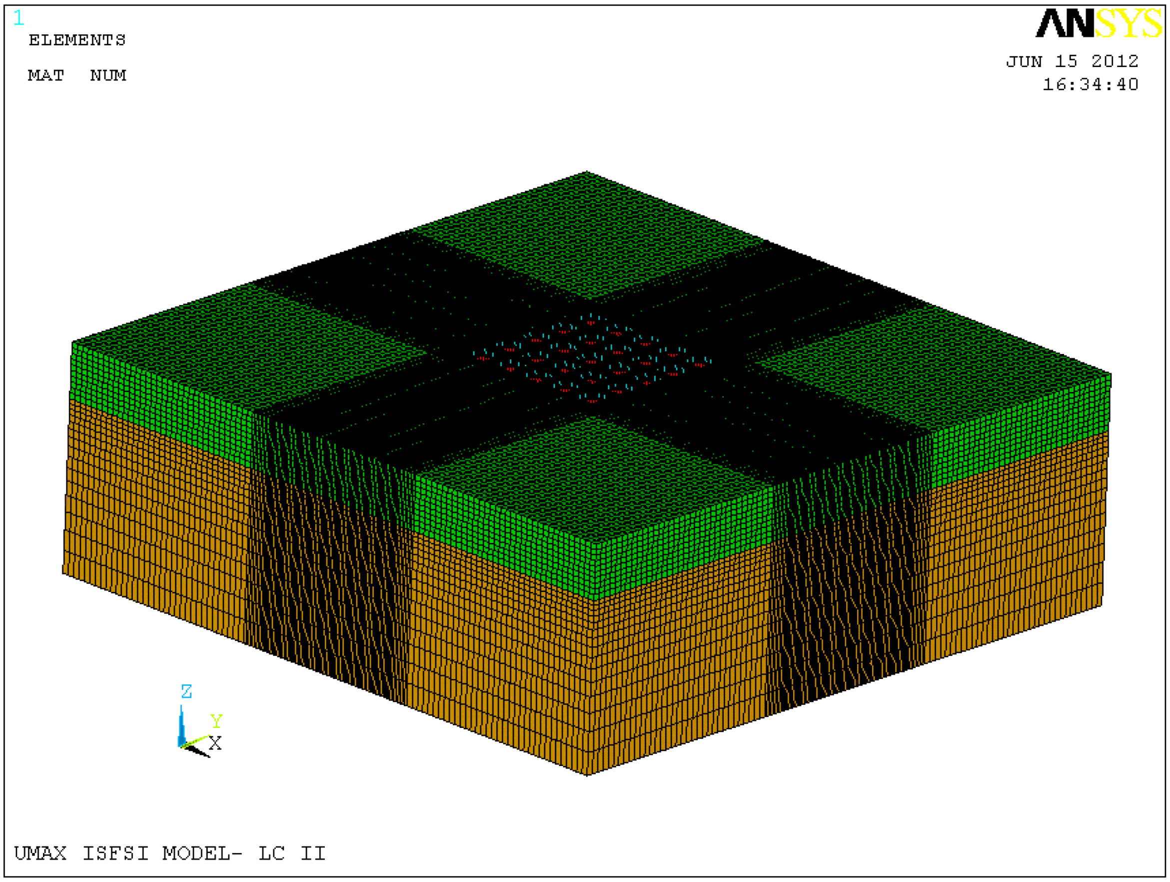


Figure 3.4.10; Finite Element Model of the ISFSI Reinforced Concrete Structures for Simulation Model II

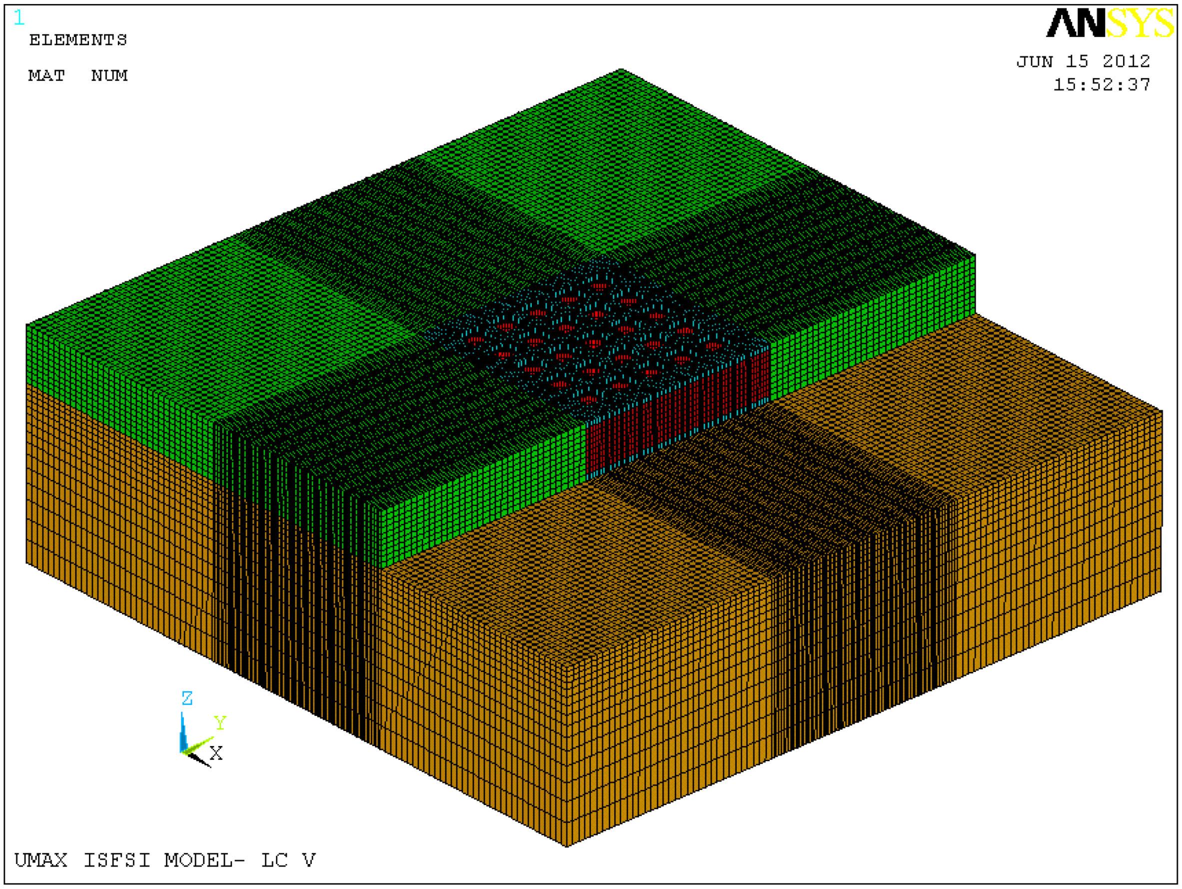
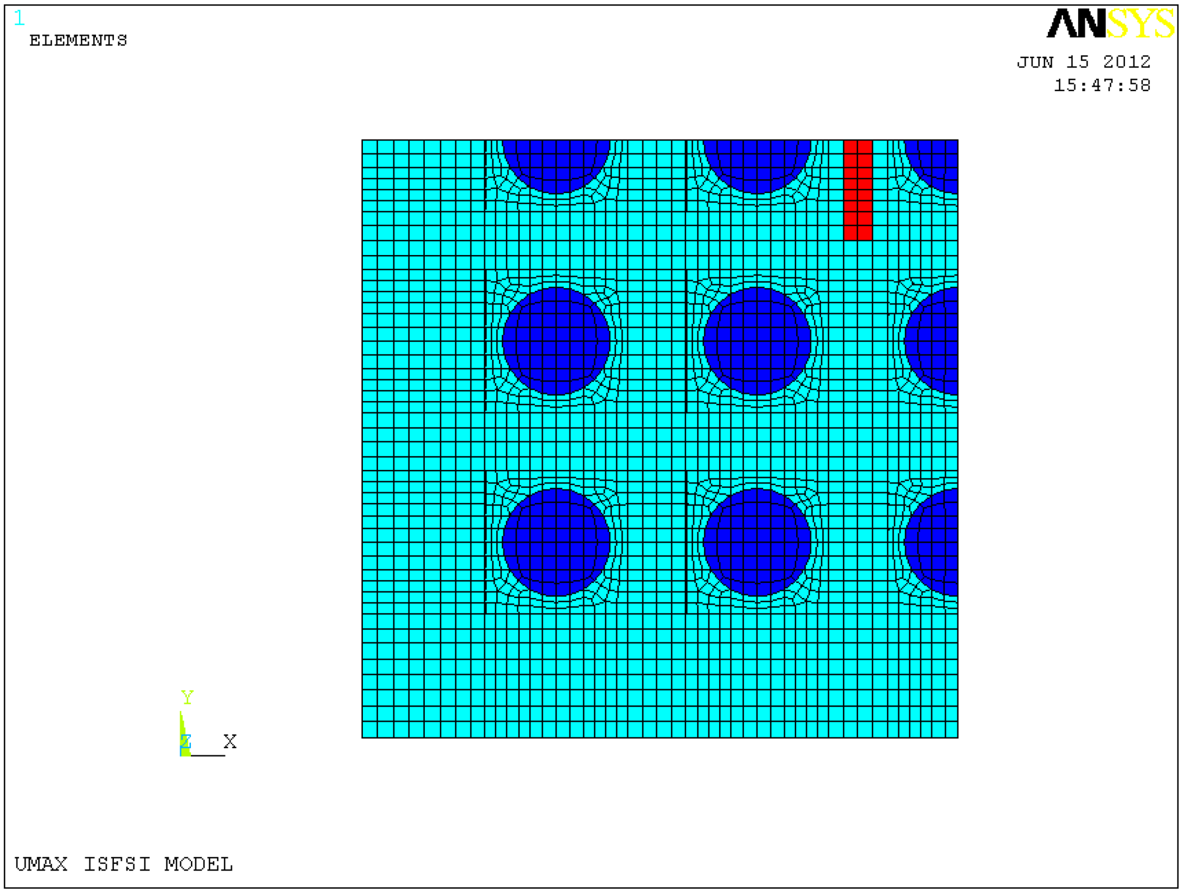
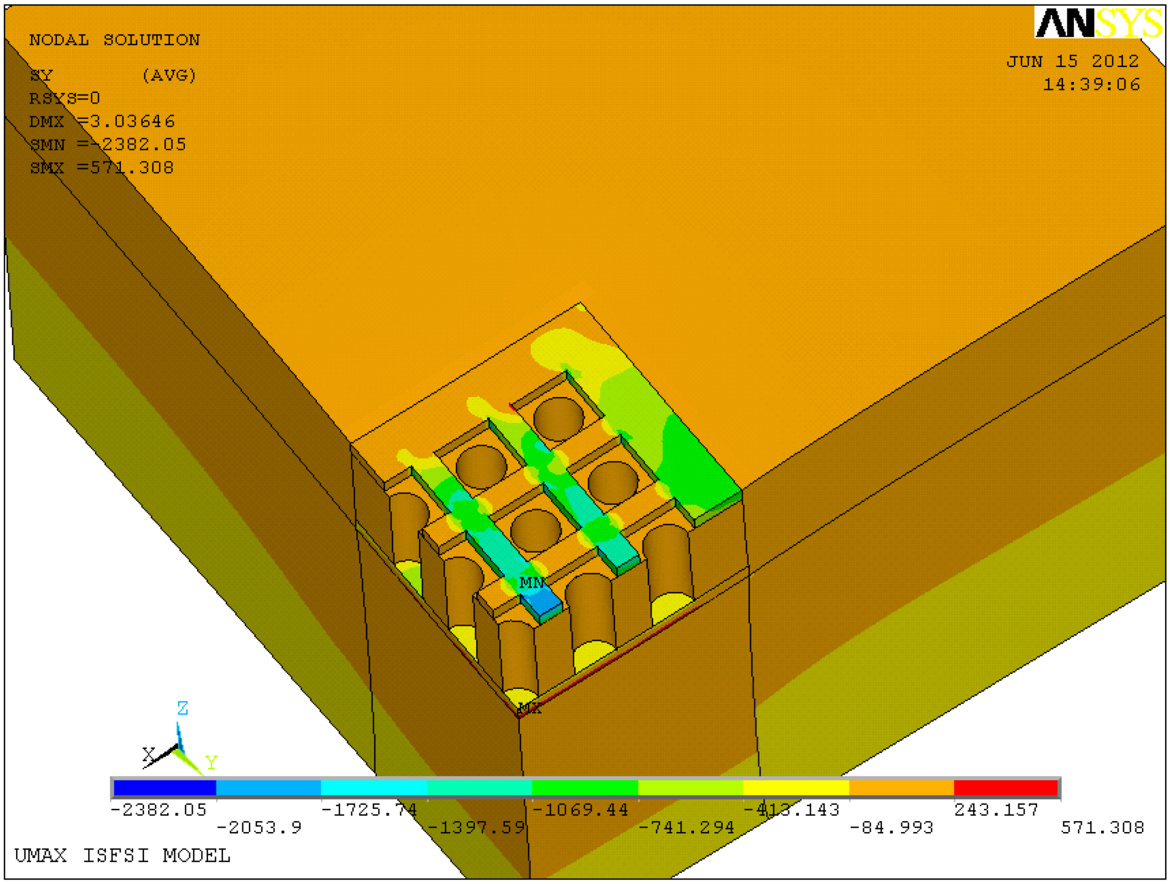


Figure 3.4.11; Finite Element Model of the ISFSI Reinforced Concrete Structures for Simulation Model V



Note: The blue footprints show the SFP area loaded with the SSC's and the red footprint represents the loaded TSP area with the transporter (VCT). The soil extending beyond the SFP boundary is not shown in the above plot for clarity.

Figure 3.4.12; ANSYS Finite Element Model of ISFSI Showing the Fully Loaded Configuration (Simulation Model I)



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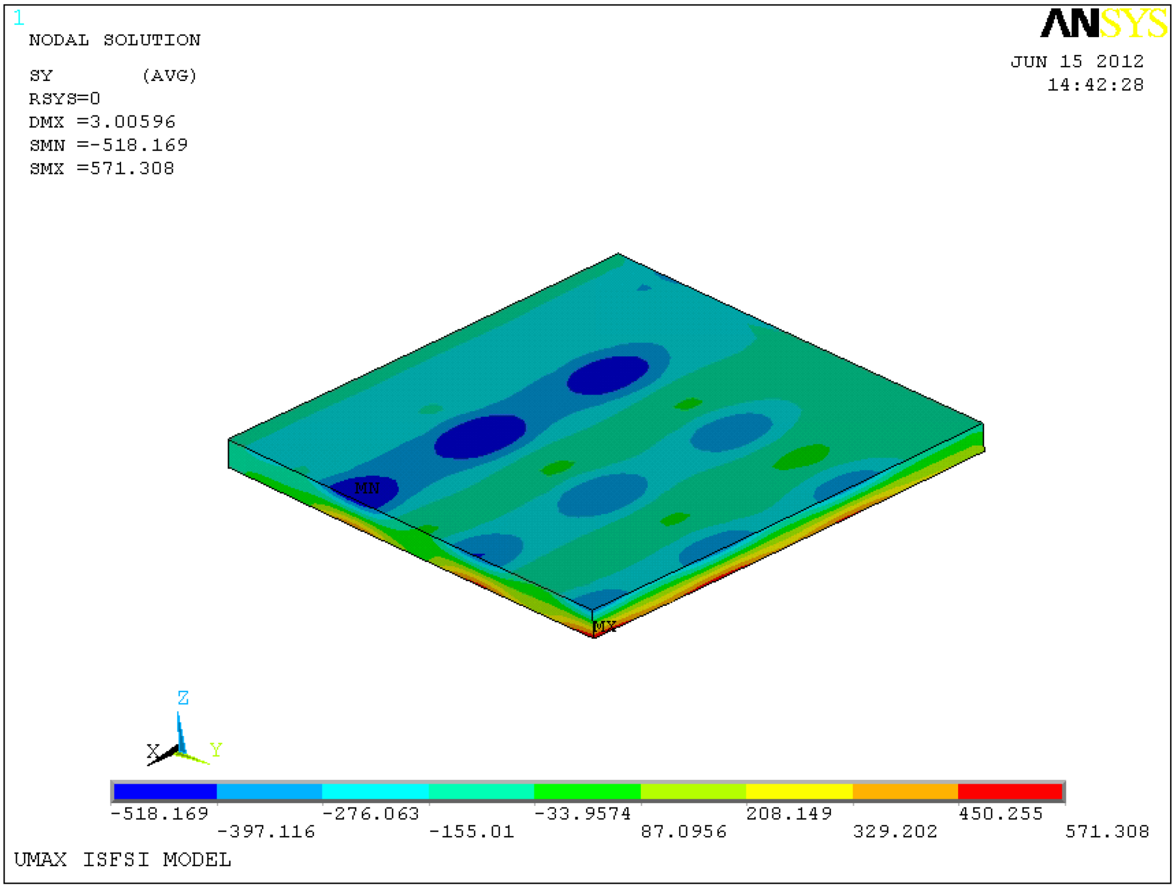

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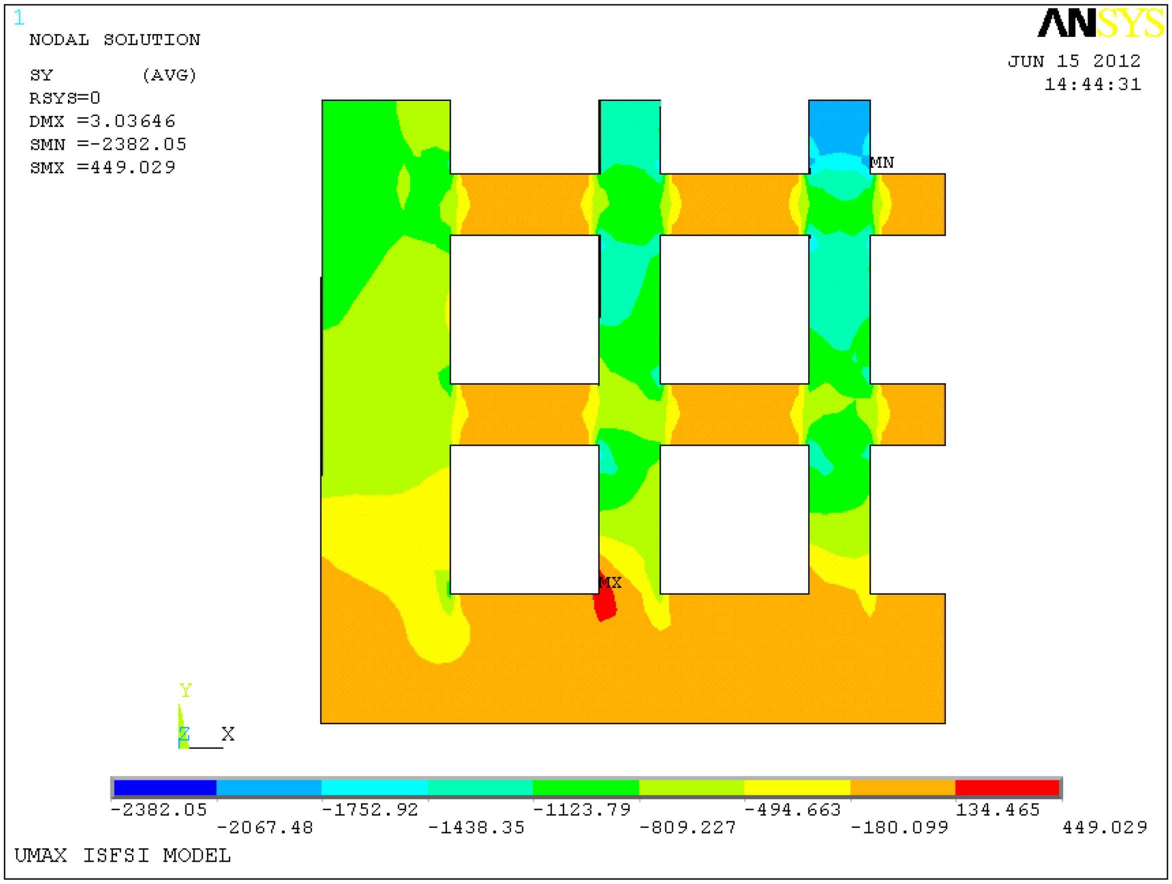
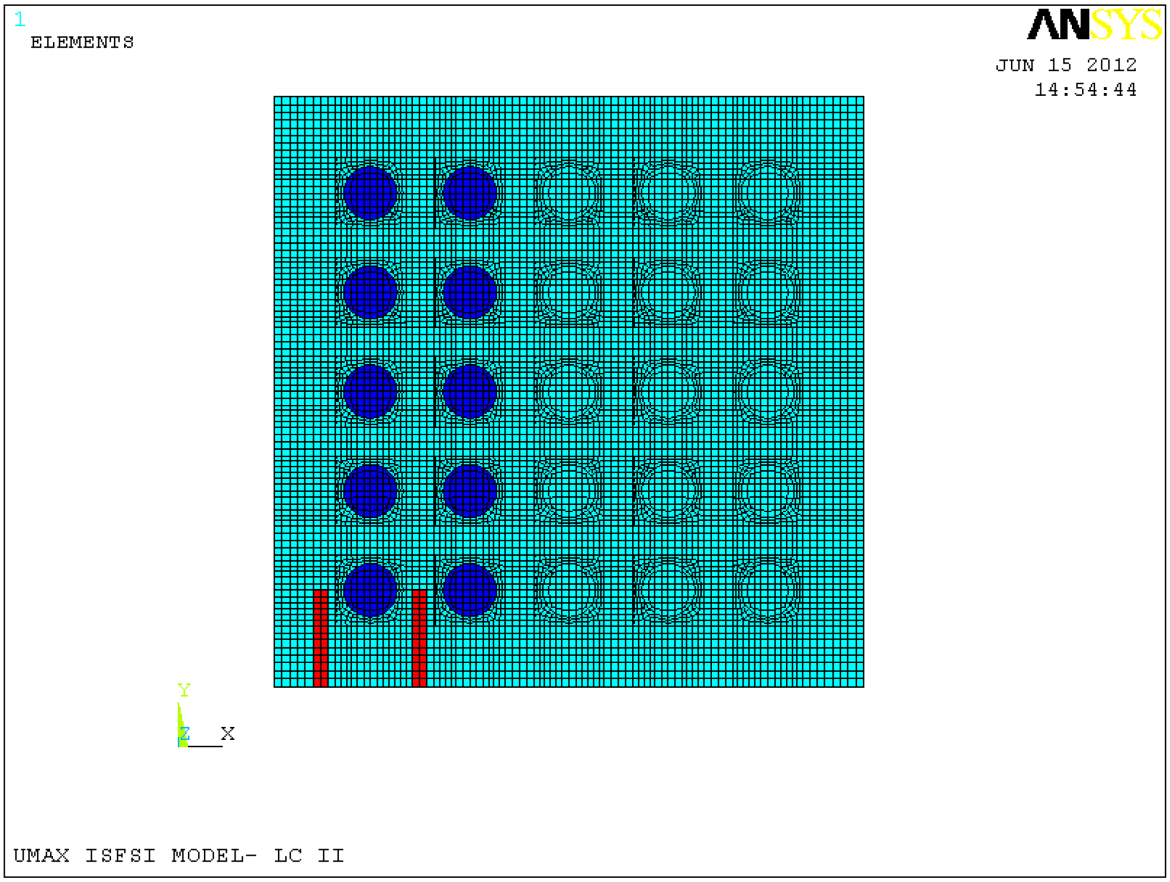


Figure 3.4.13c; Normal Stress (Sy) in ISFSI Pad Simulation Model I – Load Combination LC-3 from Table 2.4.3



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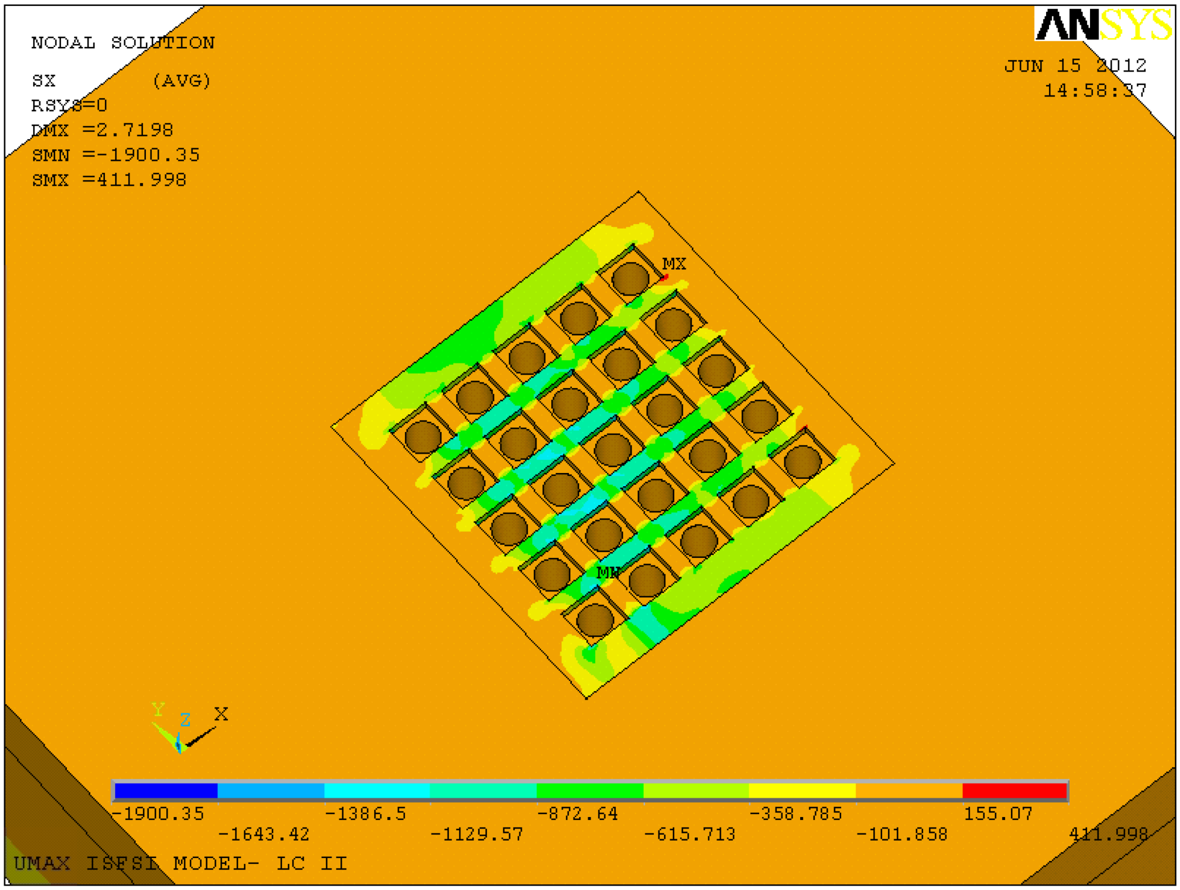

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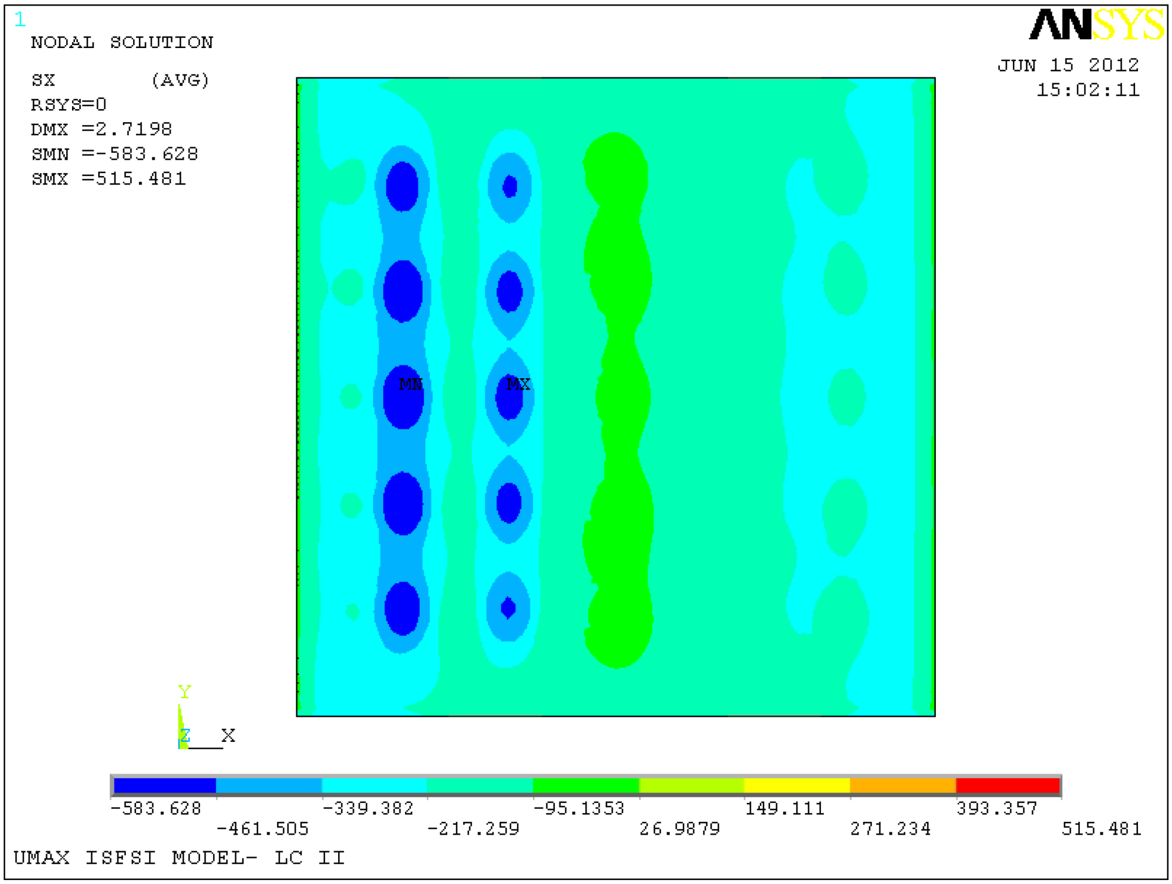


Figure 3.4.15b; Normal Stress (Sx) in SFP, Simulation Model II – Load Combination LC-3 from Table 2.4.3

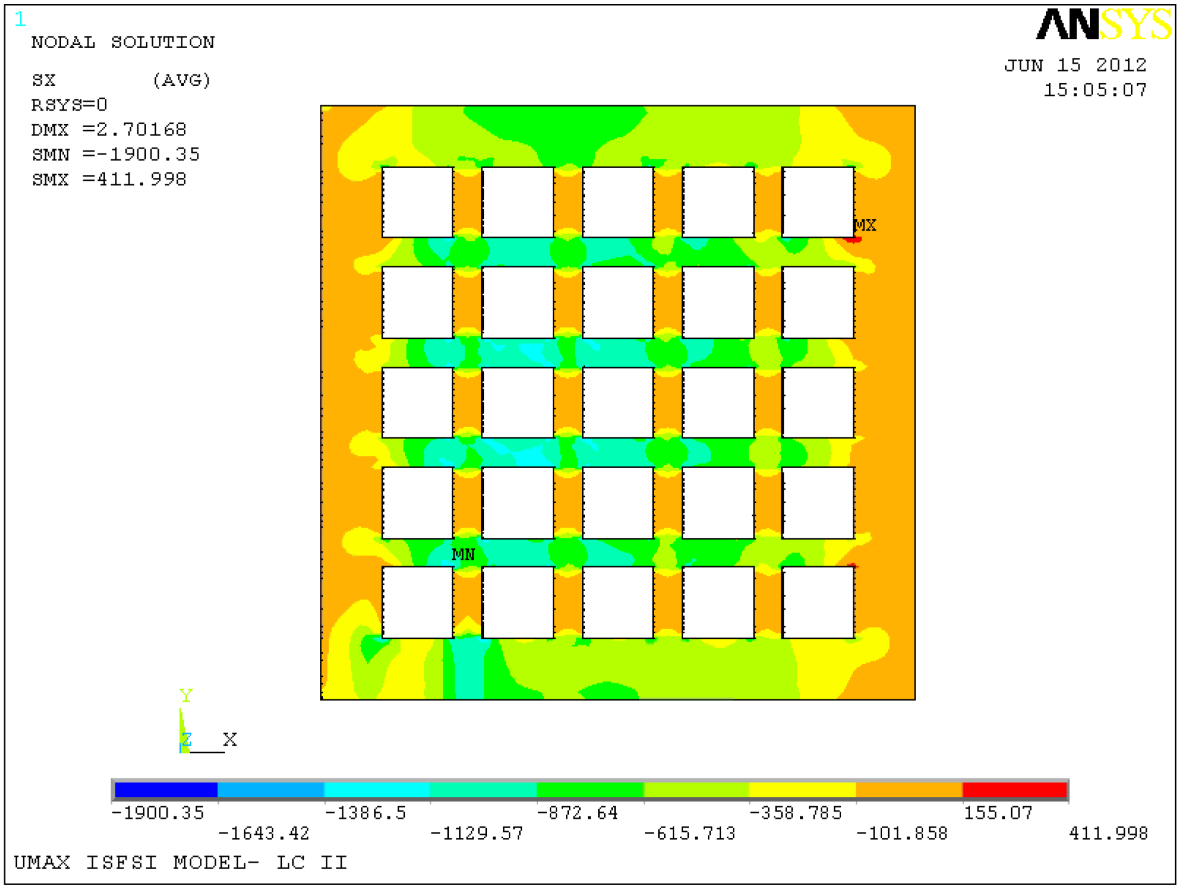


Figure 3.4.15c; Normal Stress (Sx) in ISFSI Pad, Simulation Model II – Load Combination LC-3 from Table 2.4.3

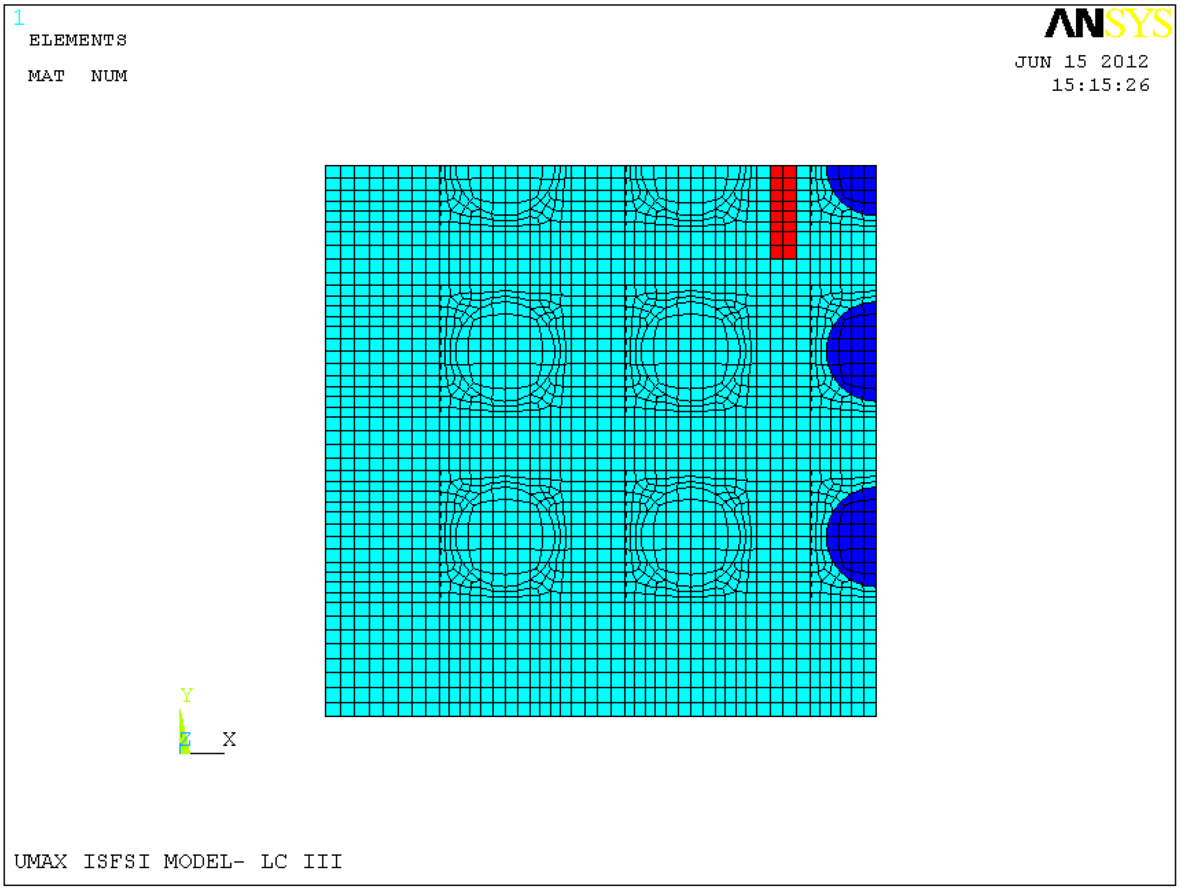


Figure 3.4.16; ANSYS Finite Element of ISFSI Showing the Center Row Loading (Simulation Model III)

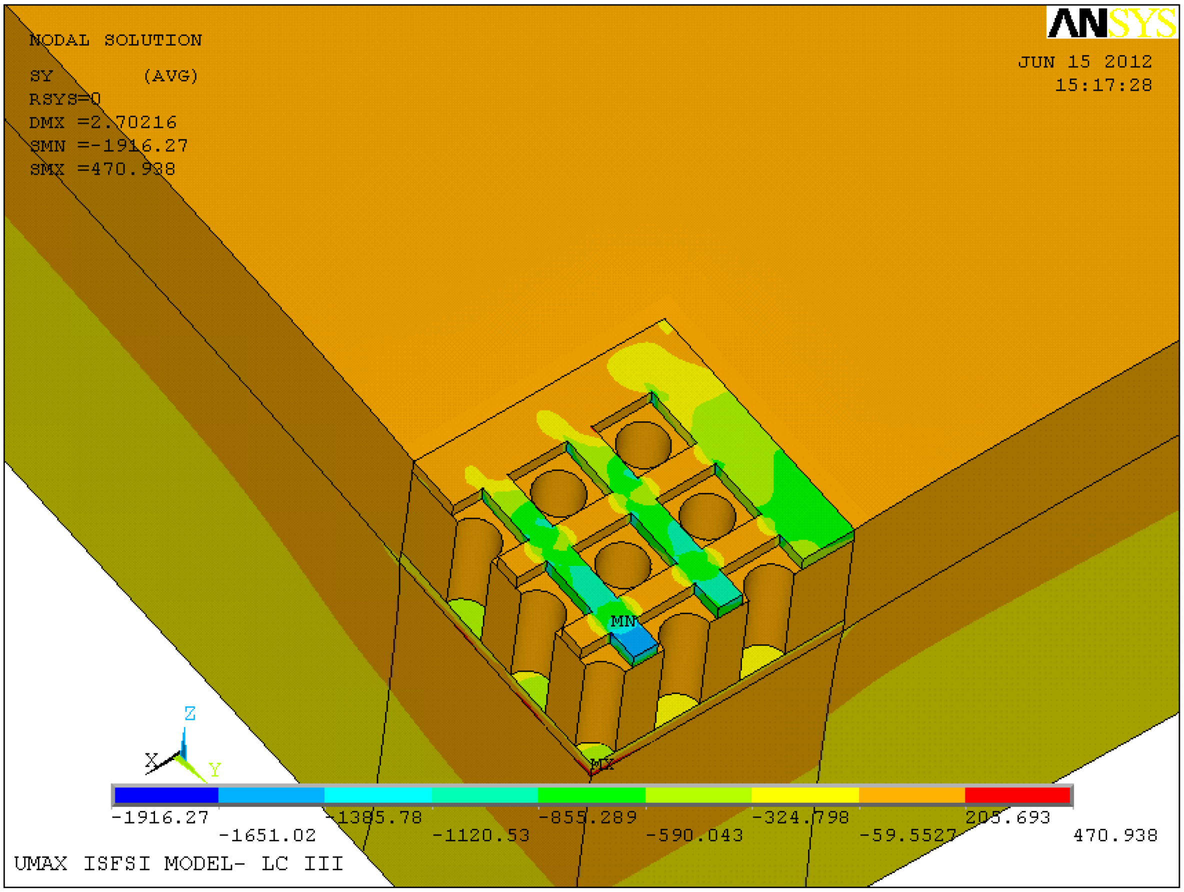


Figure 3.4.17a; Normal Stress in the ISFSI in the Direction of the Transporter Path for Simulation Model III – Load Combination LC-3 from Table 2.4.3

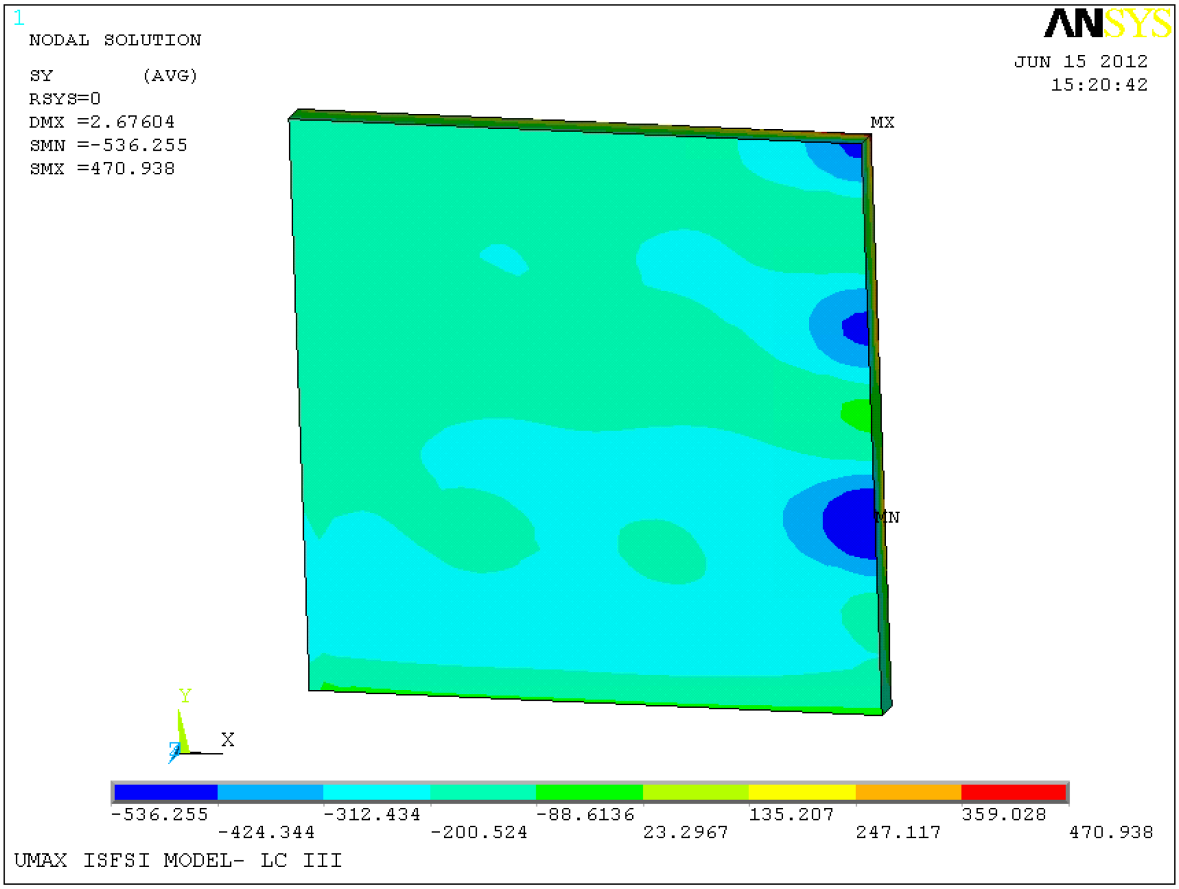
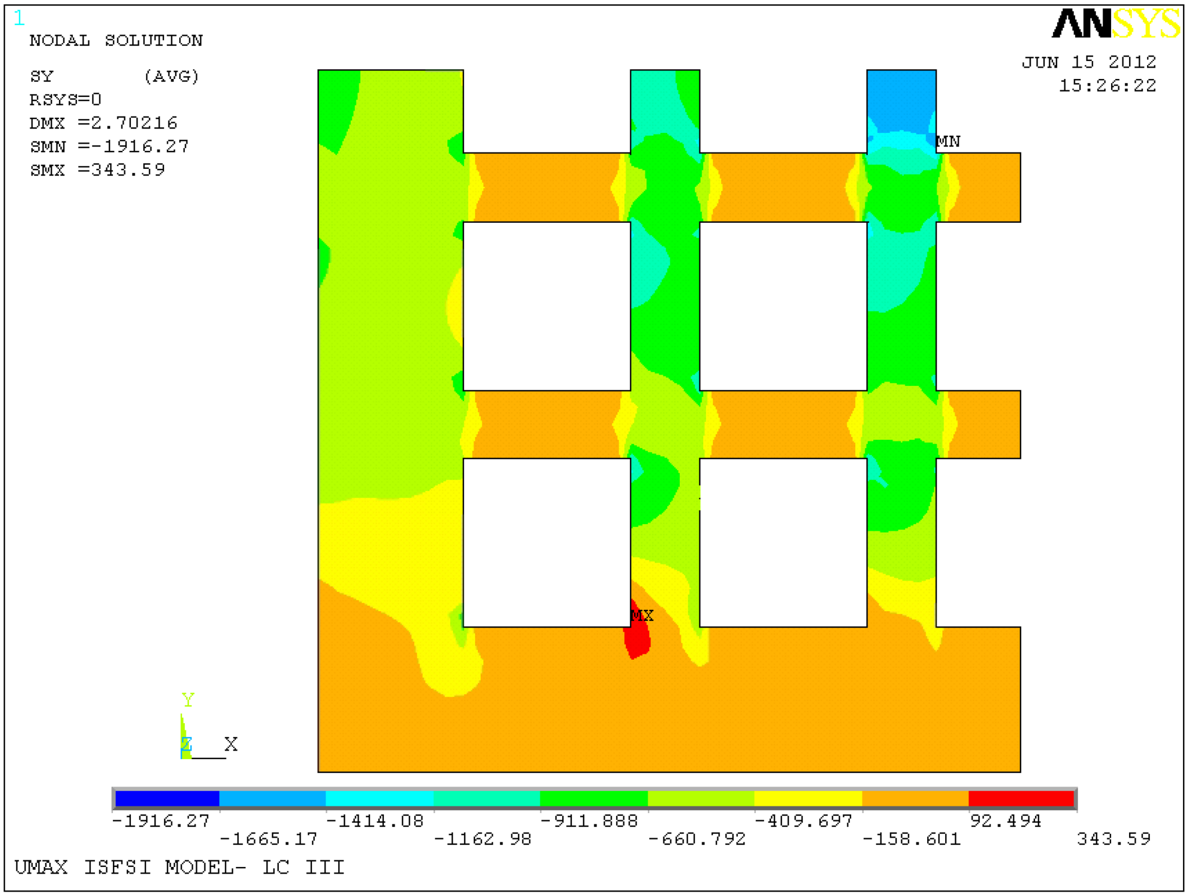


Figure 3.4.17b; Normal Stress (Sy) in SFP, Simulation Model III – Load Combination LC-3 from Table 2.4.3



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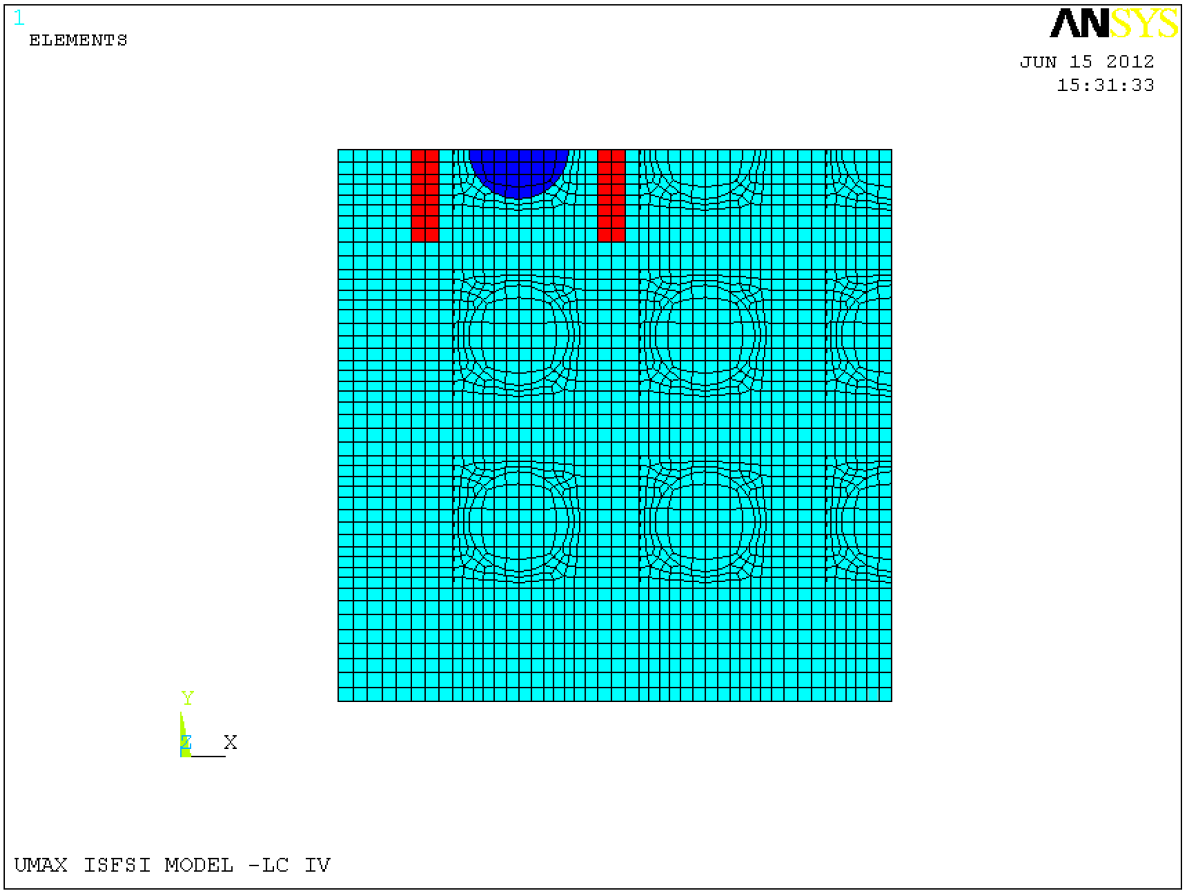

Figure 3.4.18; ANSYS Finite Element of ISFSI Showing the Single VVM Loaded (Simulation Model IV)

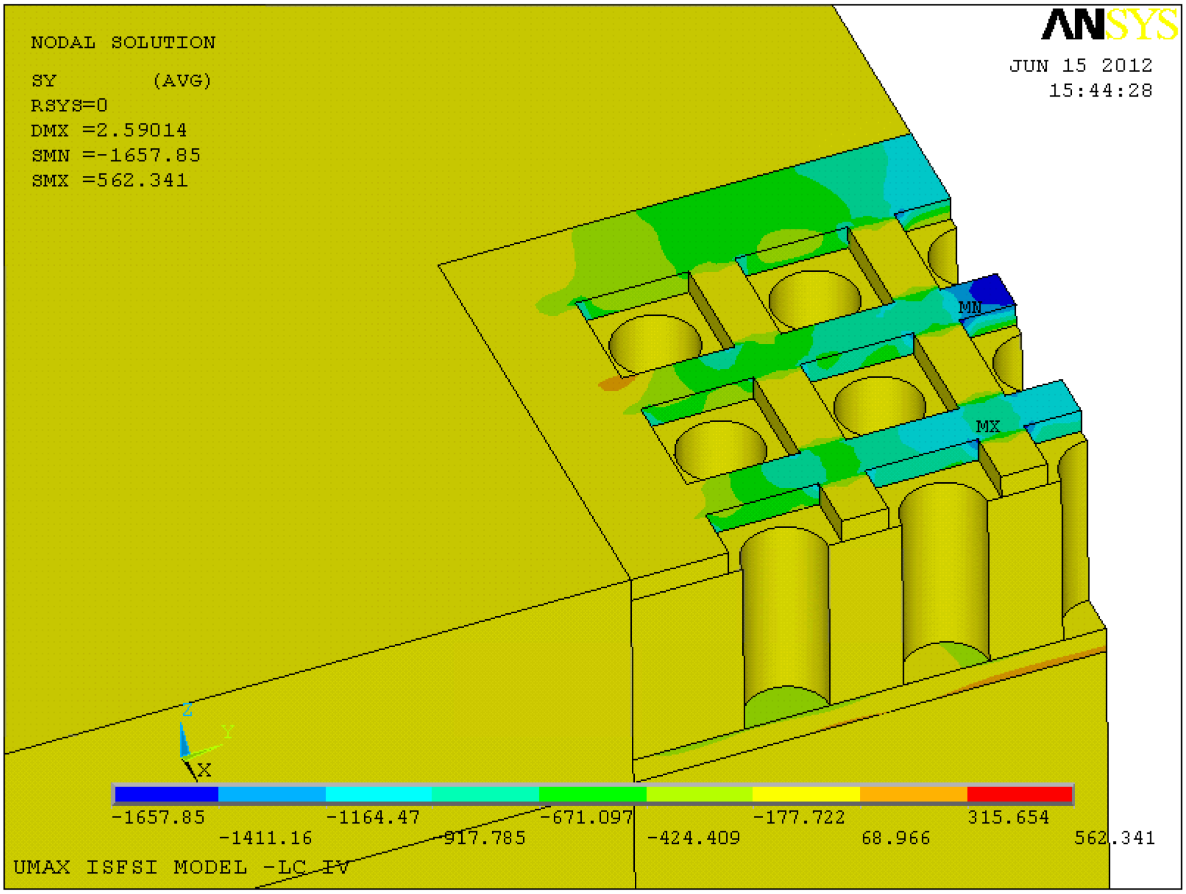

Figure 3.4.19a; Normal Stress in the ISFSI in the Direction of the Transporter Path for Simulation Model IV – Load Combination LC-3 from Table 2.4.3

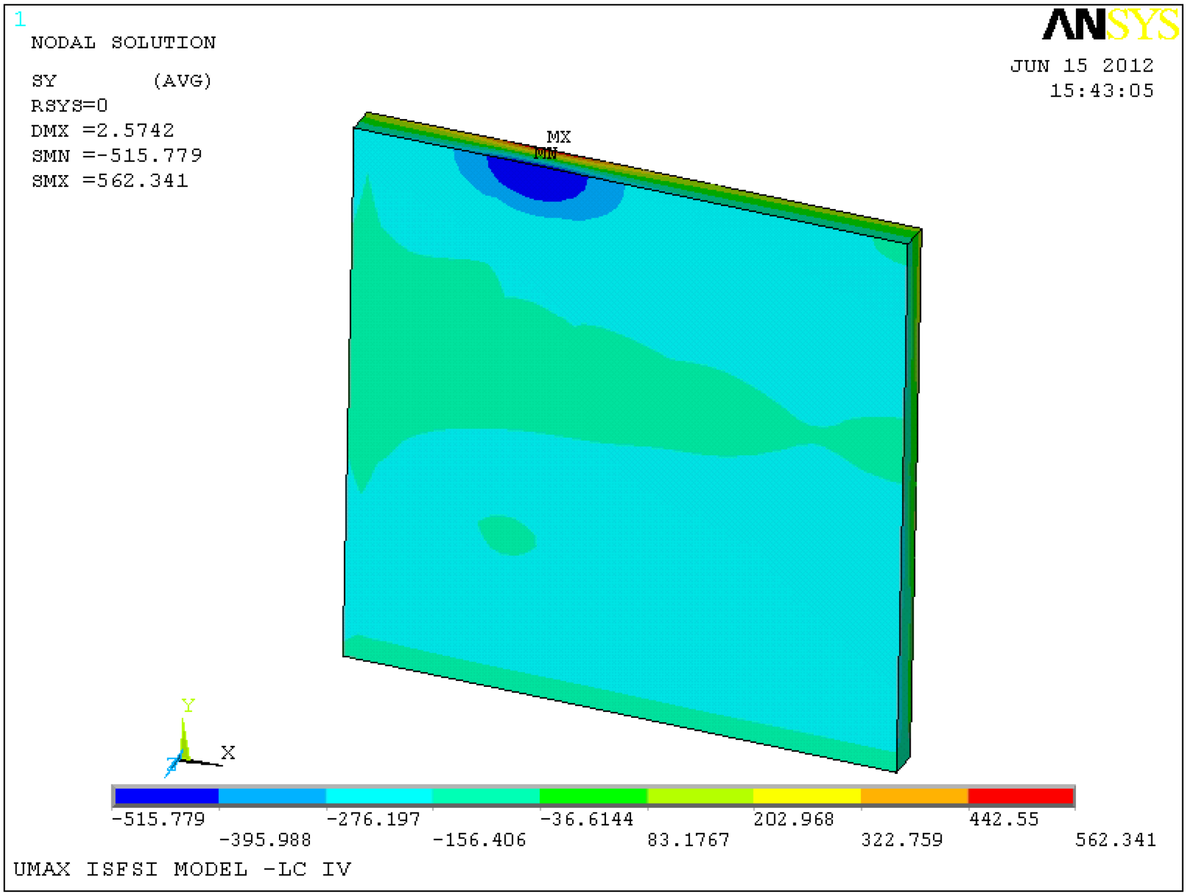

Figure 3.4.19b; Normal Stress (Sy) in SFP, Simulation Model IV – Load Combination LC-3 from Table 2.4.3

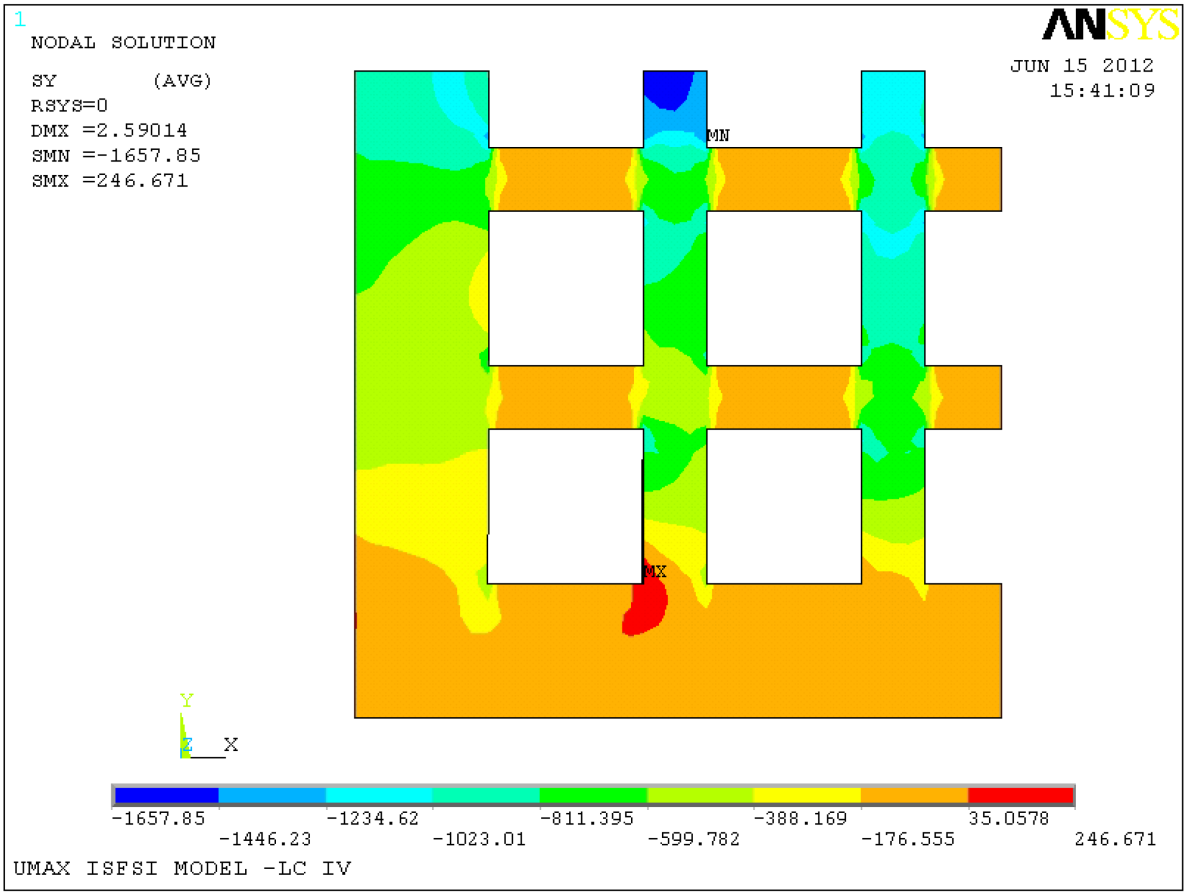

Figure 3.4.19c; Normal Stress (Sy) in ISFSI Pad, Simulation Model IV – Load Combination LC-3 from Table 2.4.3

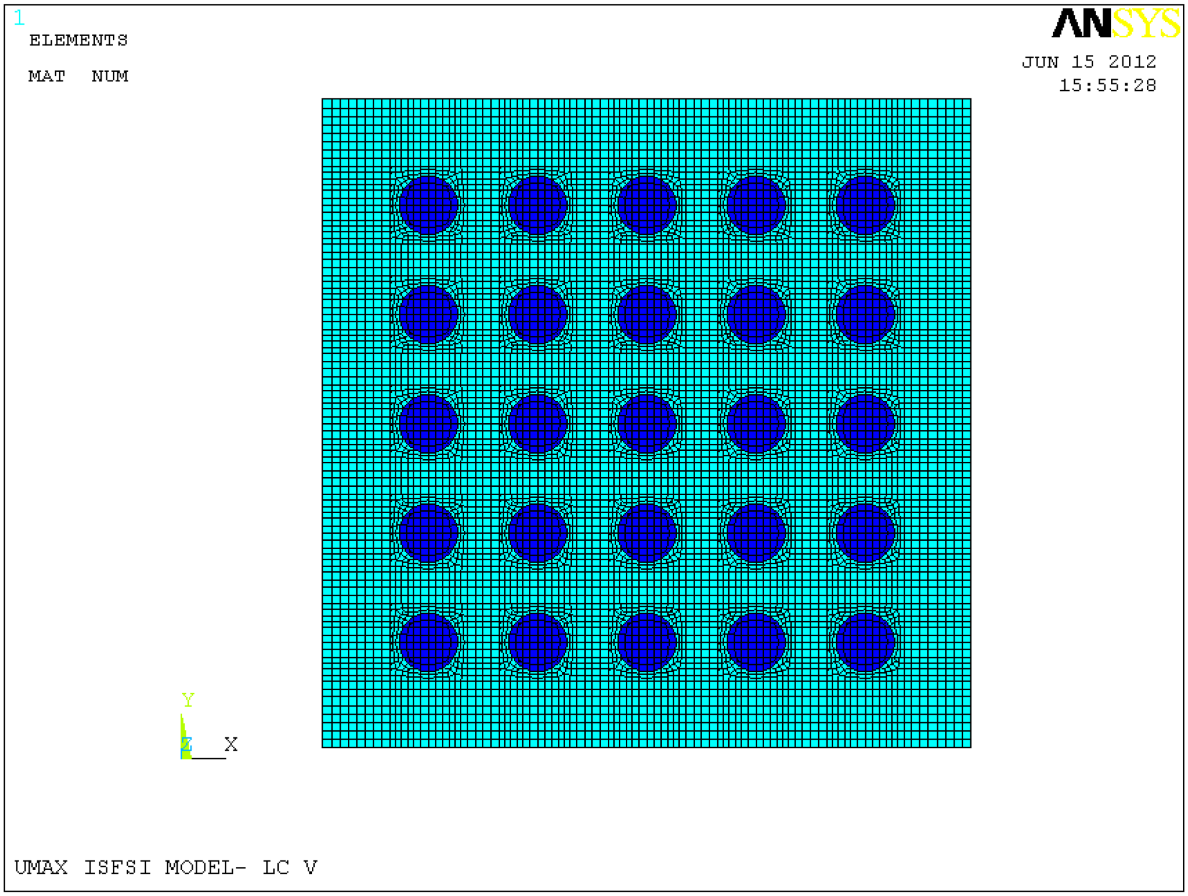

Figure 3.4.20; ANSYS Finite Element of ISFSI with fully loaded configuration (Simulation Model V)

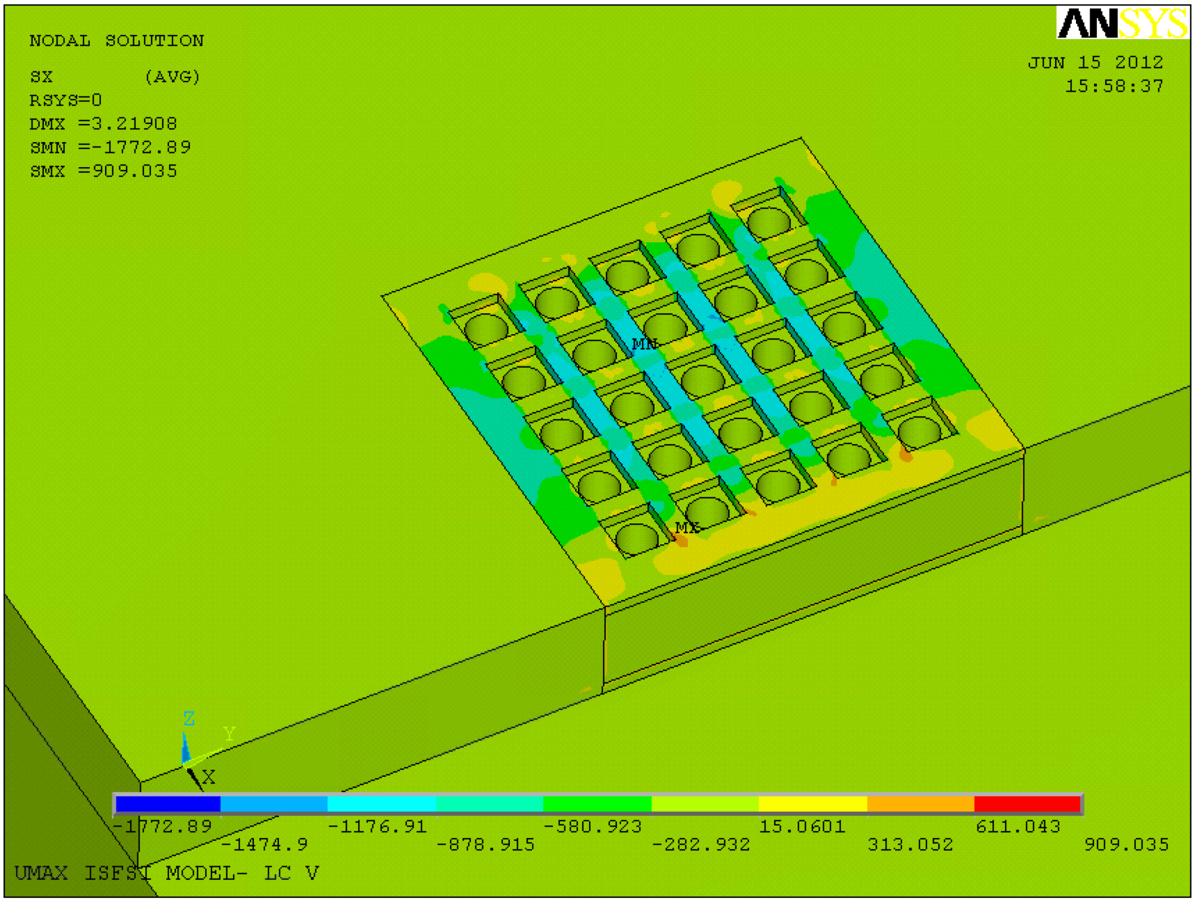

Figure 3.4.21a; Normal Stress in the ISFSI for Simulation Model V – Load Combination LC-3 from Table 2.4.3

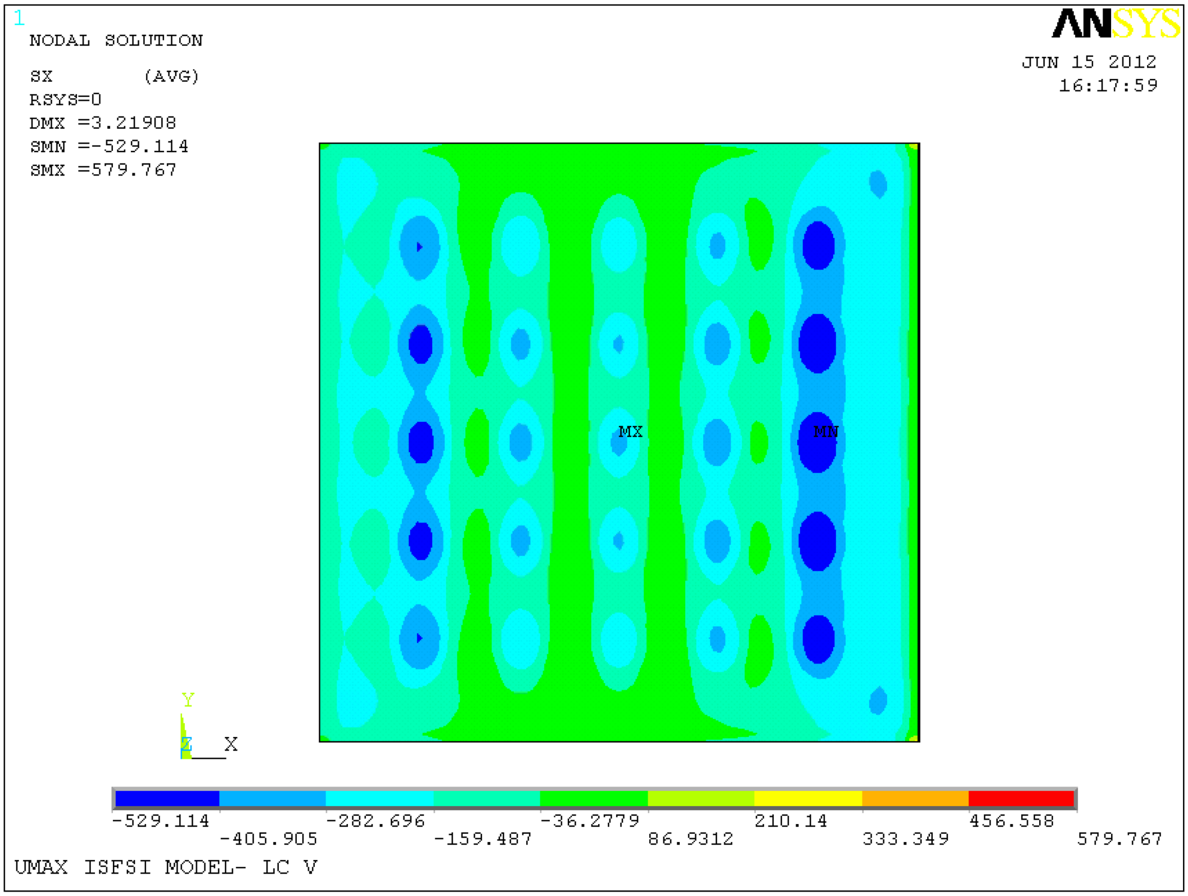

Figure 3.4.21b; Normal Stress (Sx) in SFP, Simulation Model V – Load Combination LC-3 from Table 2.4.3

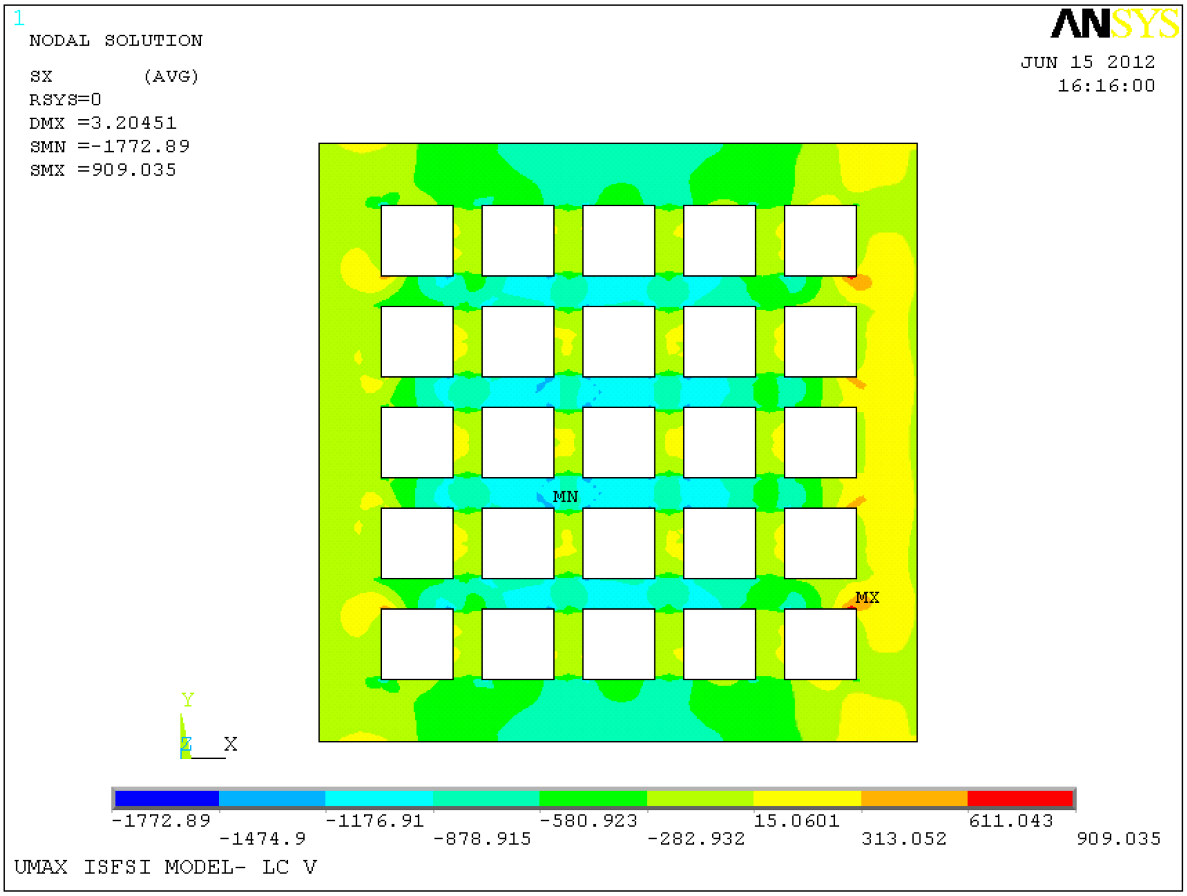

Figure 3.4.21c; Normal Stress (Sx) in ISFSI Pad, Simulation Model V – Load Combination LC-3 from Table 2.4.3

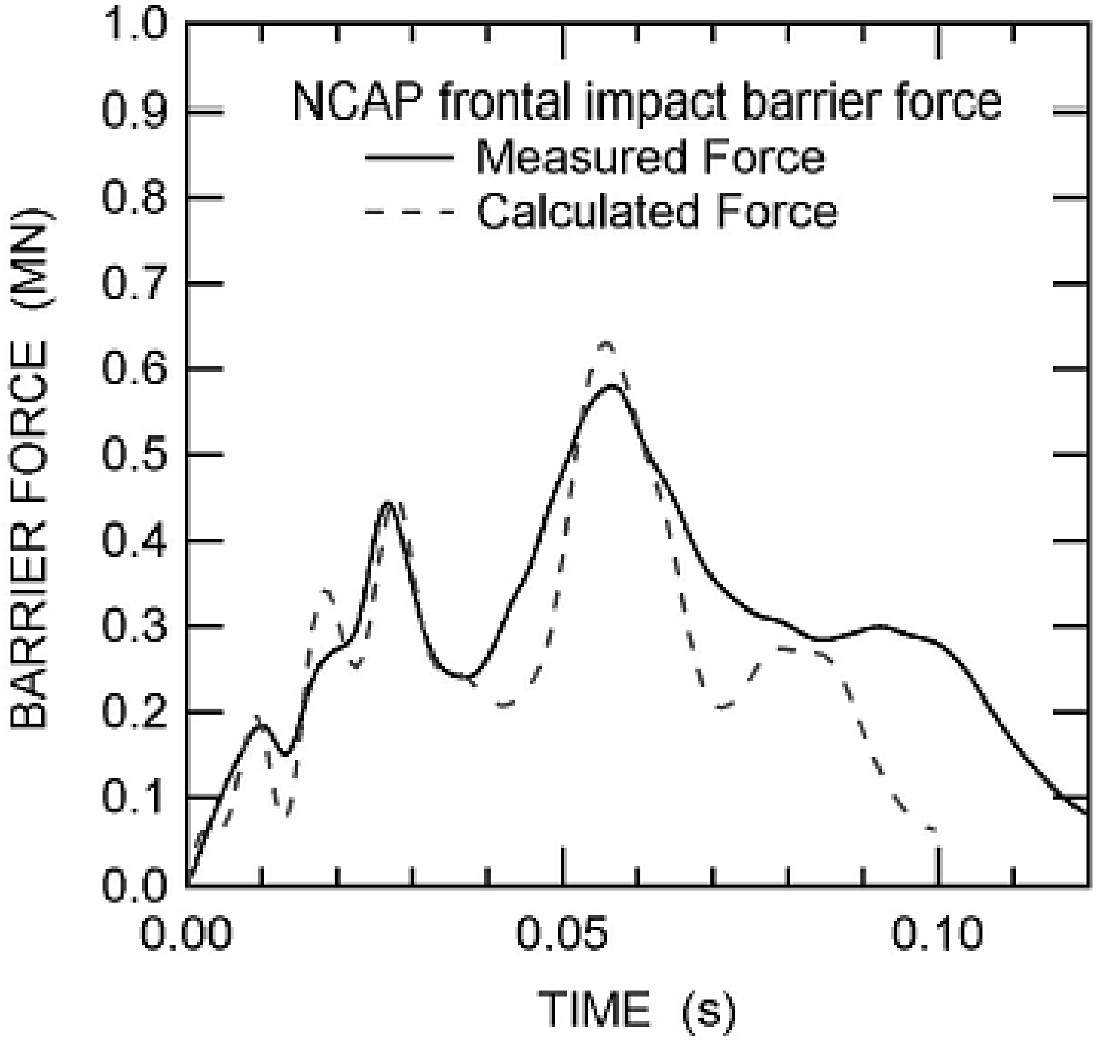

FIGURE 3.4.22; TEST RESULTS FROM 35 MPH IMPACT OF A FORD (1705 KG) AGAINST A RIGID WALL

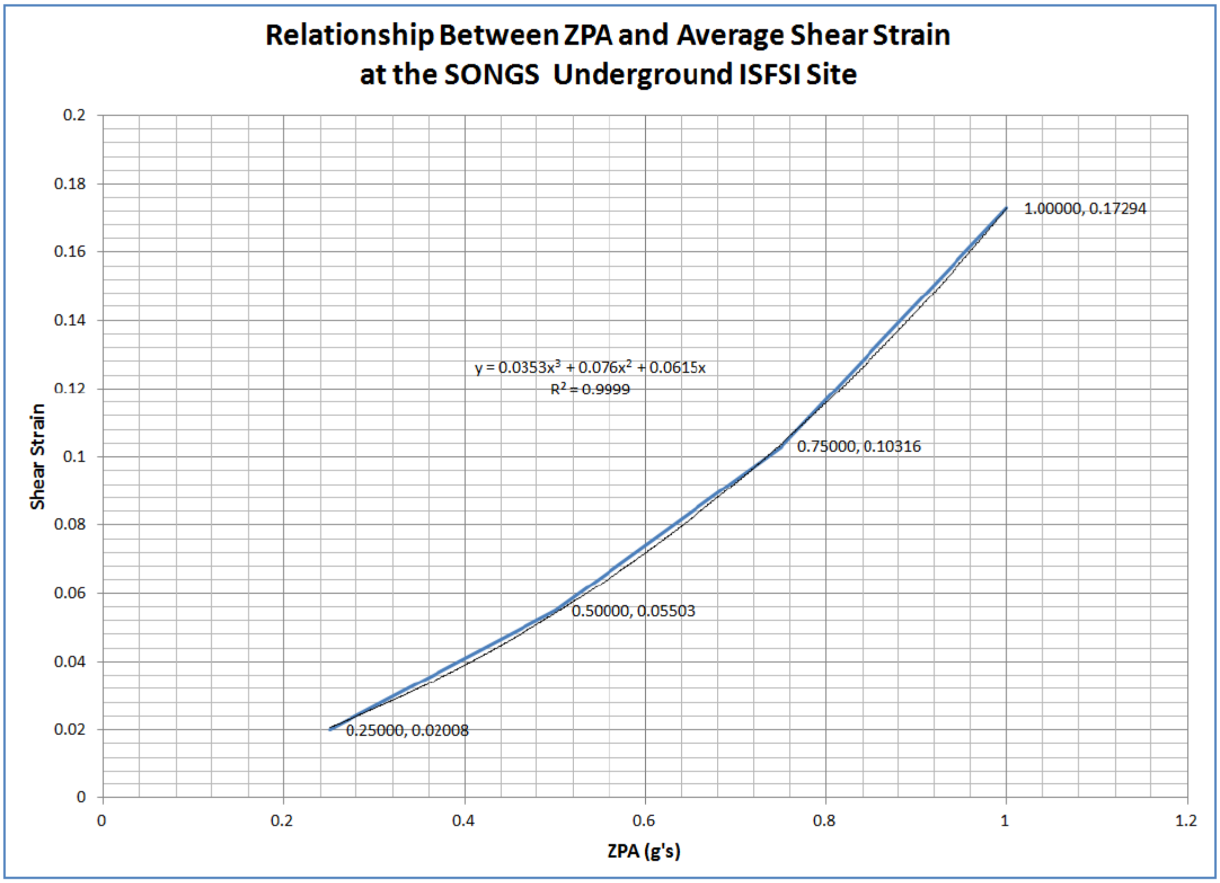

Figure 3.4.23; Relationship between ZPA and Dynamic Shear Strain

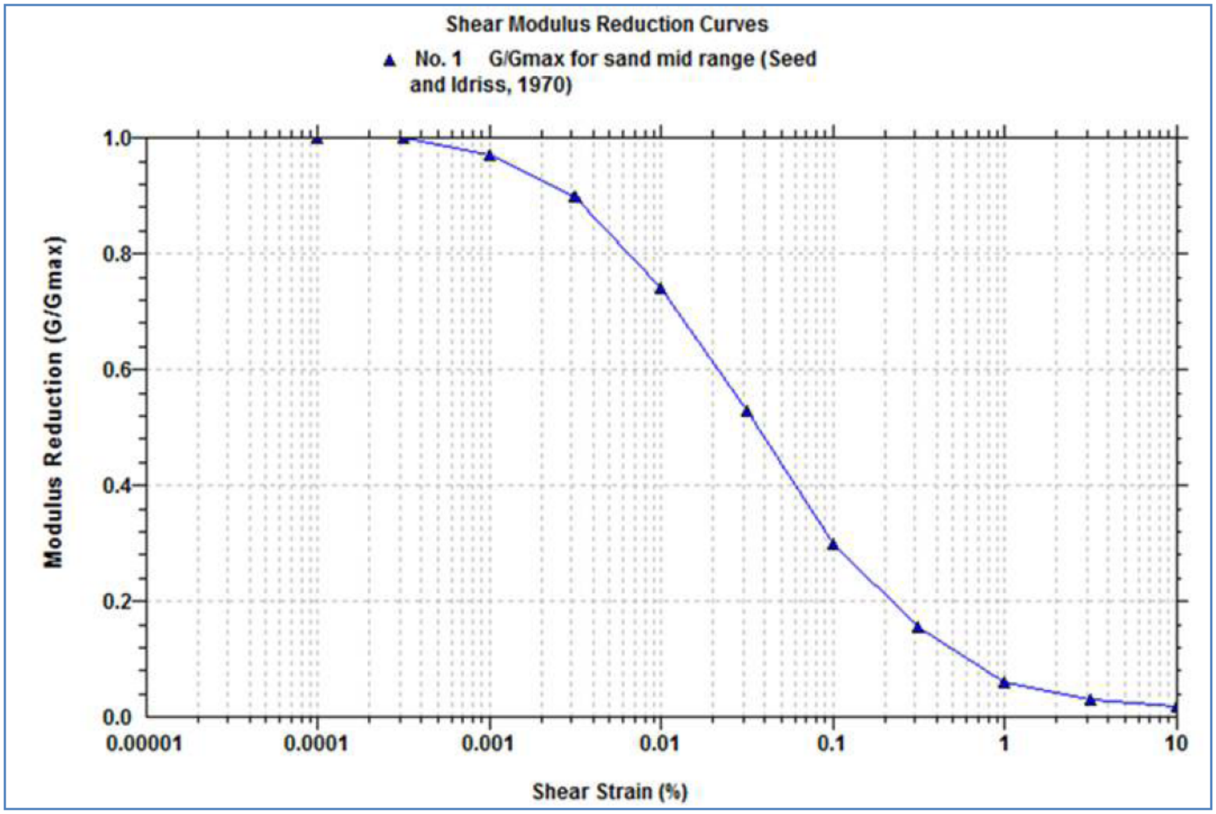

Figure 3.4.24; Relationship between Shear Strain and Shear Modulus Reduction

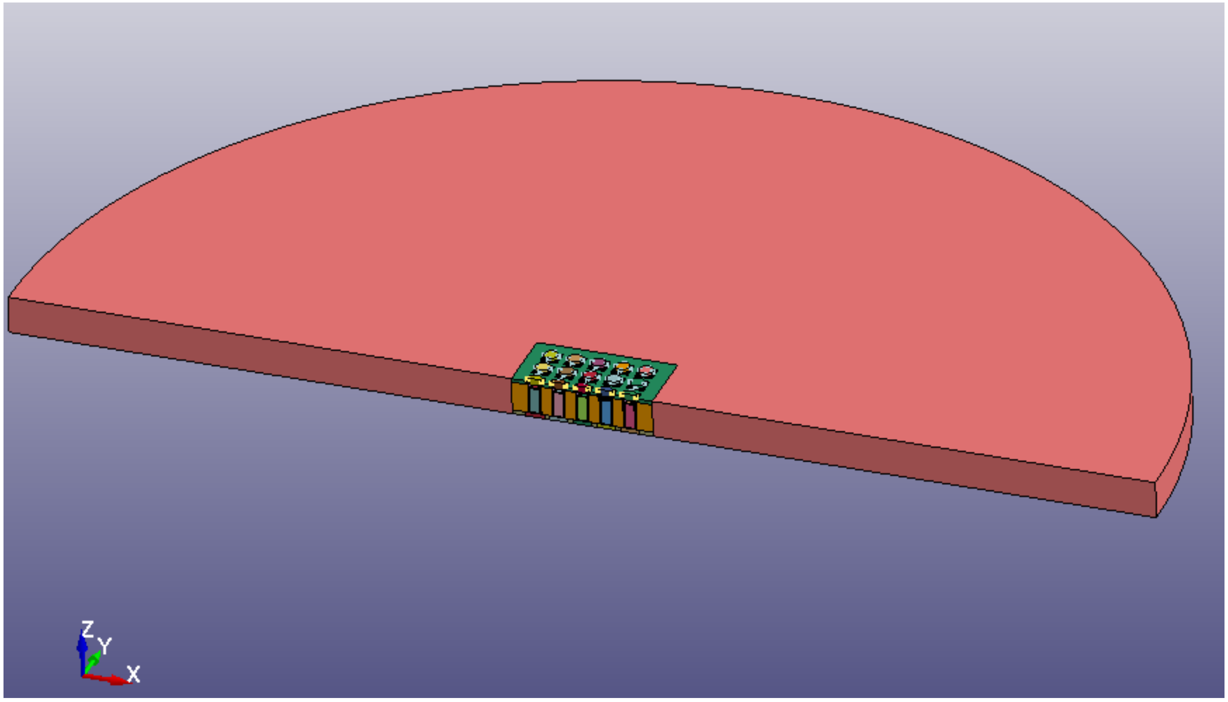

Figure 3.4.25; 3-D LSDYNA Model with "Large" Space B Subgrade for the Non-Linear SSI Analysis of the HI-STORM UMAX Version MSE

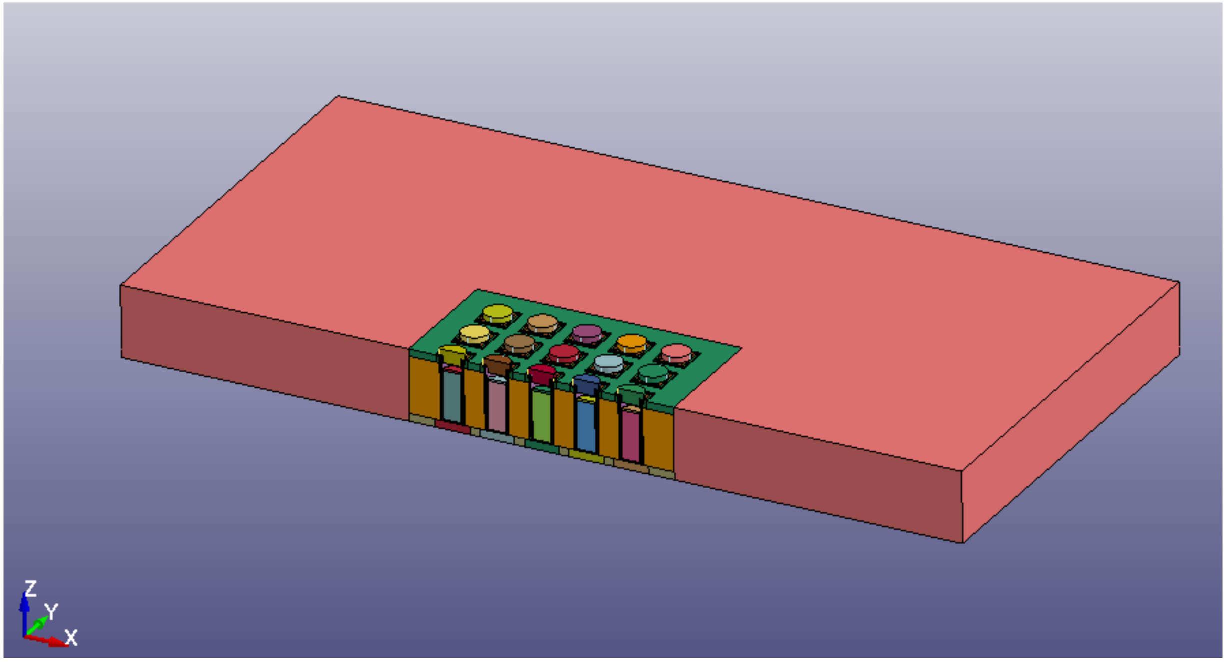

Figure 3.4.26; 3-D LSDYNA Model with "Small" Space B Subgrade for the Non-Linear SSI Analysis of the HI-STORM UMAX Version MSE

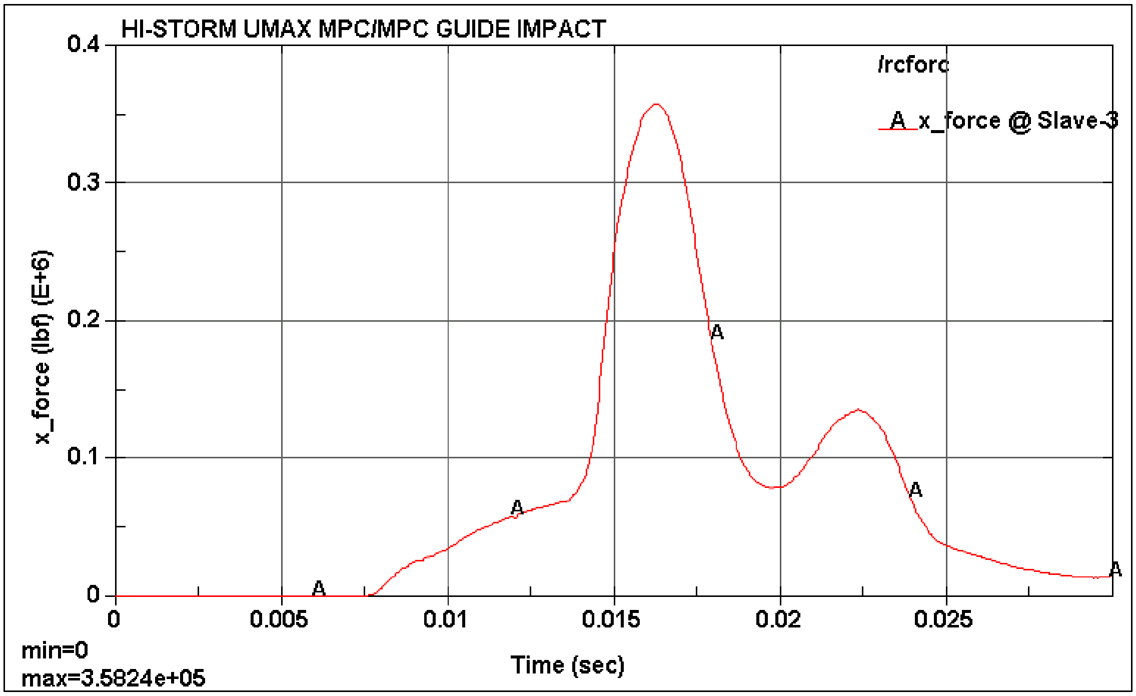

Figure 3.4.27; Time History of the Impact Force at the MPC/MPC Upper Guide Interface (Maximum force = 2×3.582×10^5 lbf = 7.164×10^5 lbf due to the half model) Obtained from the MPC to MPC Upper Guide Impact Analysis for the MSE Condition

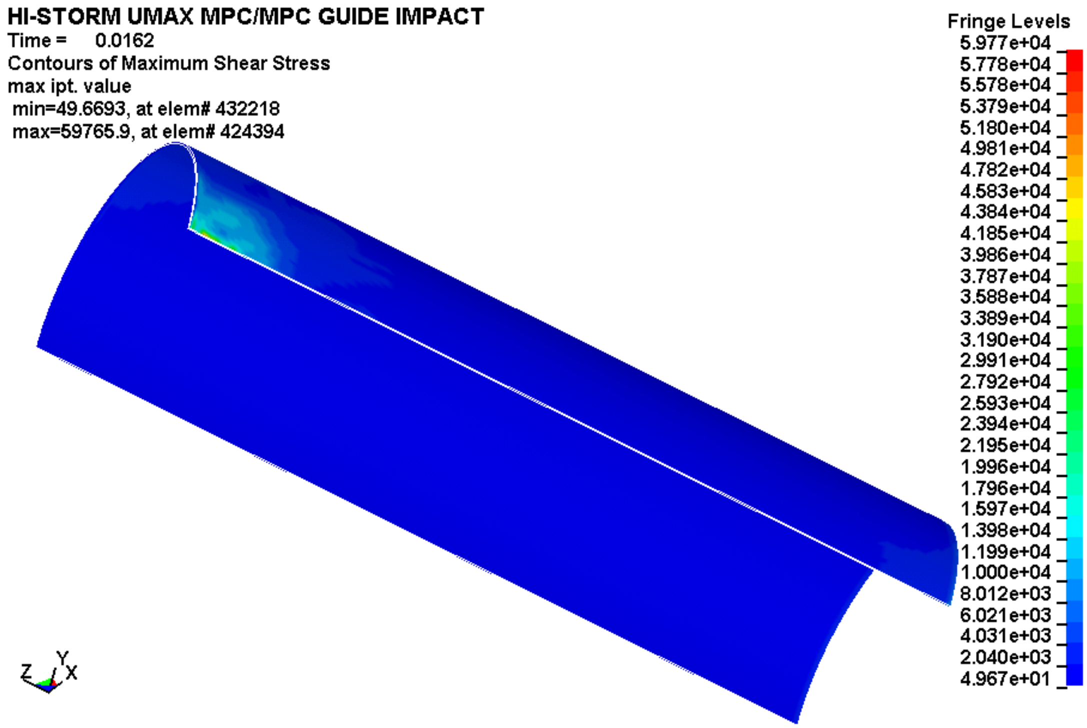

Figure 3.4.28; Maximum Shear Stress of the MPC Shell under the MSE Condition
(Maximum Primary Stress Intensity = 2×17,960 psi = 35,920 psi)

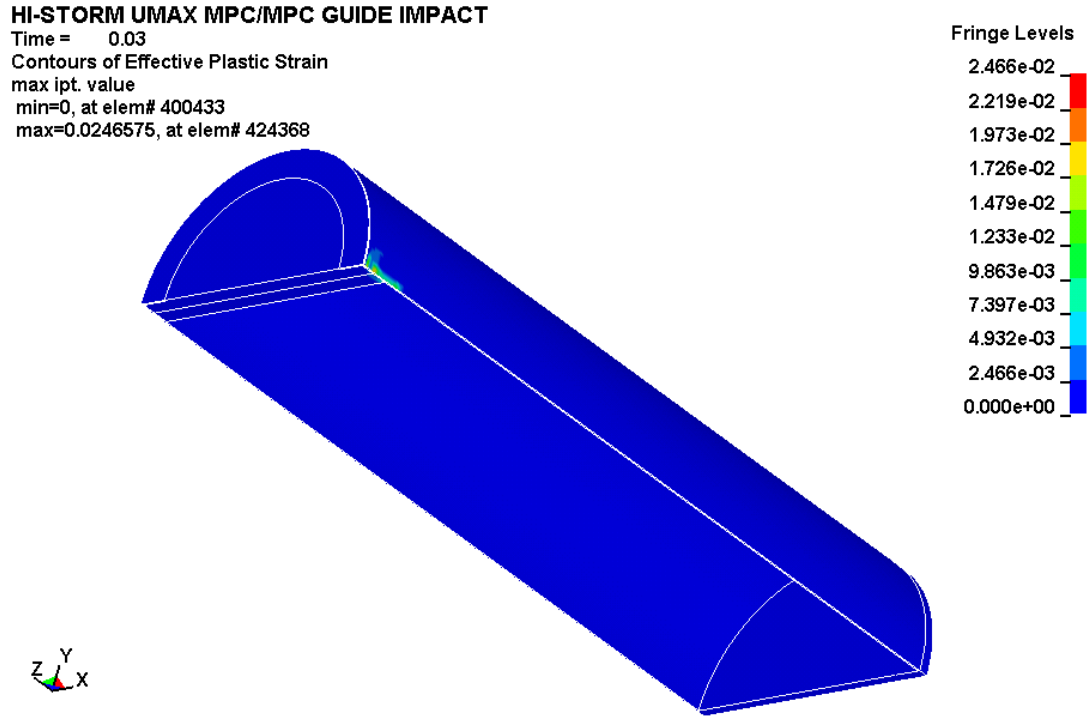

Figure 3.4.29; Maximum Plastic Strain of the MPC Enclosure Vessel Due to the Bounding Impact with the MPC Upper Guide under the MSE Condition

3.5 FUEL RODS

The regulations governing spent fuel storage cask approval and fabrication (10 CFR 72.236) require that a storage cask system "will reasonably maintain confinement of radioactive material under normal, off-normal, and credible accident conditions" (§72.236(l)). Since fuel rod cladding is not considered in the design criteria for the confinement of radioactive material under normal, off-normal, or accident conditions of storage, no specific analysis or test results for the fuel rod cladding are required to demonstrate cladding integrity.

3.6 SUPPLEMENTAL DATA

3.6.1 Calculation Packages

In addition to the calculations presented in Chapter 3, supporting calculation packages have been prepared to document other information pertinent to the analyses. Supporting calculation packages back up the summary results reported in the FSAR. The Calculation Packages are referenced in the body of the FSAR and are maintained as proprietary documents in Holtec's Configuration Control system.

3.6.2 Computer Programs

Computer programs used in this FASR are summarized in this chapter.

i. ANSYS

ANSYS is a public domain code, well benchmarked code, which utilizes the finite element method for structural analyses. It is a self contained analysis tool incorporating pre-processing (geometry creation, meshing), solver, and post processing modules in a unified graphical user interface. It can simulate both linear and non-linear material and geometric behavior. It includes contact algorithms to simulate surfaces making and breaking contact, and can be used for both static and dynamic simulations. ANSYS has been independently QA validated by Holtec International and used for structural analysis of casks, fuel racks, pressure vessels, and a wide variety of SSCs, for over twenty years.

ii. LS-DYNA

LS-DYNA is a general purpose finite element code for analyzing the large deformation static and dynamic response of structures including structures coupled to fluids. The main solution methodology is based on explicit time integration and is therefore well suited for the examination of the response to shock loading. A contact-impact algorithm allows difficult contact problems to be easily treated. Spatial discretization is achieved by the use of four node tetrahedron and eight node solid elements, two node beam elements, three and four node shell elements, eight node solid shell elements, truss elements, membrane elements, discrete elements, and rigid bodies. A variety of element formulations are available for each element type. Adaptive re-meshing is available for shell elements. LS-DYNA currently contains approximately one hundred constitutive models and ten equations-of-state to cover a wide range of material behavior.

In this safety analysis report, LS-DYNA is used to analyze all loading conditions that involve short-time dynamic effects.

LS-DYNA is maintained in a QA-validated status in Holtec's Configuration Control system.

iii. Shake 2000 [3.4.4]

SHAKE 2000, QA validated in [3.6.1] under Holtec International's quality program, has been used to characterize the transformation of the components of an earthquake as they traverse through layers of soil. SHAKE 200 is a linear ,elastic wave propagation code based on a 1972 report by P.B. Schnabel, Lysmer, J. and H.B. Seed at the Earthquake Engineering Research Center entitled "SHAKE: A Computer Program for Earthquake analysis of Horizontally Layered Sites", EERC report number 71-12. SHAKE 2000 has been used in this FSAR to define the earthquake response spectra at the TOG and at the SFP levels consistent with the assumed properties of the subgrade and the undergrade.

The above mentioned computer codes have been benchmarked and QA-validated to establish their veracity. The compliance matrix in Table 3.6.1 below provides the necessary information to document their validation status, and the measures employed pursuant to ISG-21 and Holtec's QA program, to ensure error-free solutions.

Table 3.6.1

ISG-21 AND QA COMPLIANCE MATRIX FOR COMPUTER CODES

	Item	ANSYS	LS-DYNA	Shake 2000
1.	Benchmark and QA-validation are documented in Holtec Report No.(s) (Proprietary Reports)	HI-2012627	HI-961519	HI-2022827 HI-2104792
2.	Computer Program Type (Public or Private Domain)	Public Domain	Public Domain	Public Domain
3.	Does Holtec maintain a system evaluating error notices if any are issued by the Code provider to evaluate their effect on the safety analyses carried out using the Code, including Part 21 notification? (Yes/No)	Yes	Yes	Yes
4.	Is the use of the Code restricted to personnel qualified under the Company's personnel qualification program? (Yes/No)	Yes	Yes	No
5.	Has benchmarking been performed against sample problems with known independently obtained numerical solutions (Yes/No)	Yes	Yes	Yes
6.	Have element types used in the safety analyses herein also employed in the benchmarking effort? (Yes/No)	Yes	Yes	N/A
7.	Are the element types used in this FSAR also used in other Holtec dockets that support other CoCs? (Yes/No)	Yes	Yes	N/A
8.	Is each update of the Code vetted for backwards consistency with prior updates? (Yes/No)	Yes	Yes	Yes
9.	Is the use of the Code limited to the range of parameters specified in the User Manual provided by the Code Developer? (Yes/No)	Yes	Yes	Yes
10.	Are the element aspect ratios, where applicable, used in the simulation model within the limit recommended by the Code Developer or Holtec's successful experience in other safety analyses? (Yes/No)	Yes	Yes	N/A
11.	Are element sizes used in the simulation models consistent with past successful analyses in safety significant applications? (Yes/No)	Yes	Yes	N/A
12.	Was every computer run in this chapter free of an error warning (i.e., in hidden warnings in the Code that indicate a possible error in the solution? (Yes/No)	Yes	Yes	Yes
13.	If the answer to the above is No, then is the annotated warning discussed in the discussion of the result in this report?	N/A	N/A	N/A

3.7 COMPLIANCEWITH THE STRUCTURAL REQUIREMENTS IN PART 72

Supporting information to provide reasonable assurance with respect to the adequacy of the HI-STORM UMAX system to store spent nuclear fuel in accordance with the stipulations of 10CFR72 is presented throughout this FSAR. The following statements are applicable to an affirmative structural safety evaluation:

- The design and structural analysis of the HI-STORM UMAX System is in full compliance with the provisions of Chapter 3 of NUREG-1536 as appropriate for a vertical ventilated module assembly (see Table 3.7.1).

- The HI-STORM UMAX structures, systems, and components (SSC) that are important to safety (ITS) are identified in the Licensing Drawings in Section 1.5. The Licensing Drawings present the HI-STORM UMAX SSCs in adequate detail and the explanatory narratives in this chapter provide sufficient textual details to allow an independent evaluation of their structural effectiveness.

- The requirements of 10CFR72.24 with regard to information pertinent to structural evaluation are provided in Chapters 2, 3, and 12.

- Technical Specifications pertaining to the structures of the HI-STORM UMAX system have been provided in Chapter 13 herein pursuant to the requirements of 10CFR72.26.

- A series of analyses to demonstrate compliance with the requirements of 10CFR72.122(b) and (c), and 10CFR72.24(c)(3) have been performed which show that SSCs in the HI-STORM UMAX VVM designated as ITS possess an adequate margin of safety with respect to all load combinations applicable to normal, off-normal, accident, and natural phenomenon events. In particular, the following information is provided:

    i. Load combinations for the HI-STORM UMAX VVM and the ISFSI structures for normal, off-normal, accident, and natural phenomenon events have been provided.

    ii. Stress limits applicable to the Code materials can be found in Section 3.3.

    iii. The stress and displacement response of the MPC Enclosure vessel, and the HI-STORM UMAX VVM for all applicable loads, have been computed by analysis and reported in Subsections 3.4.3 and 3.4.4. Descriptions of stress analysis models are presented in Subsection 3.1.3.

- The criticality safety of the stored MPCs is ensured by demonstrating that the maximum g-load sustained by the MPC stored inside the HI-STORM UMAX VVMs is less than their licensing basis value. This conclusion satisfies the requirement of 10CFR72.124(a), with respect to structural margins of safety for SSCs important to nuclear criticality safety.

- Structural margins of safety during handling, packaging, and transfer operations, under the provisions of 10CFR Part 72.236(b), imply that the lifting and handling devices be engineered to comply with the stipulations of ANSI N14.6, NUREG-0612. The requirements of the governing standards for handling operations are summarized in Subsection 3.4.3 herein. Factors of safety for all ITS components under lifting and handling operations are summarized in tables in Section 3.4, which show that adequate structural margins exist in all cases.

- Consistent with the requirements of 10CFR72.236(i), the confinement boundary for the HI-STORM UMAX System has been engineered to maintain confinement of radioactive materials under normal, off-normal, and postulated accident conditions. This assertion of confinement integrity is made on the strength of the MPC's licensing basis in docket number 72-1032 and the following information provided in this FSAR.

- The information on structural design included in this FSAR complies with the requirements of 10CFR72.120 and 10CFR72.122.

- The structural design features in the HI-STORM UMAX VVM (along with the previously certified MPCs) are in compliance with the specific requirements of 10CFR72.236(e), (f), (g), (h), (i), (j), (k), and (m).

Table 3.7.1

NUREG-1536 COMPLIANCE MATRIX FOR 10CFR72.120 AND 10CFR72.122 REQUIREMENTS

Item	Compliance	Notes
i. Design and fabrication to acceptable quality standards	All ITS components designed and fabricated to recognized Codes and Standards: • Enclosure Vessel: Subsection NB, loc. cit. • HI-STORM UMAX Structure: Subsection NF, loc. cit.	Subparagraph 3.1.2.3
ii. Erection to acceptable quality standards	• Concrete in HI-STORM UMAX closure lid meets requirements of: ACI –318 (2005)	Subsection 3.3.2
iii. Testing to acceptable quality standards	• All non-destructive examination of structural components utilizes Section V of the ASME Code. • Testing for radiation containment per provisions of NUREG-1536 • Concrete testing	Chapter 8
iv. Adequate structural protection against environmental conditions and natural phenomena.	Analyses presented in Chapter 3 demonstrate that the confinement boundary will preserve its integrity under all postulated off-normal and natural phenomena events listed in Chapter 2.	Subparagraph 3.4.4.1 Chapter 11
v. Adequate protection against fires and explosions	• The extent of combustible (exothermic) material in the vicinity of the cask system is procedurally controlled (the sole source of hydrocarbon energy is diesel in the tow vehicle). • Analyses show that the heat energy released from the postulated fire	Subsections 12.3.20 and 12.3.21 Subsection 11.2.4

Table 3.7.1

NUREG-1536 COMPLIANCE MATRIX FOR 10CFR72.120 AND 10CFR72.122 REQUIREMENTS

Item	Compliance	Notes
	accident condition surrounding the cask will not result in impairment of the confinement boundary and will not lead to structural failure of the VVM. The effect on shielding will be localized to the external surfaces directly exposed to the fire without a significant change in HI-STORM UMAX VVM. • Explosion effects are shown to be bounded by the Code external pressure design basis and there is no adverse effect on ready retrievability of the MPC.	Subparagraph 3.4.4.1
vi. Appropriate inspection, maintenance, and testing	Inspection, maintenance, and testing requirements set forth in this FSAR are in full compliance with the governing regulations and established industry practice.	See Chapters 1 and 2.
vii. Adequate accessibility in emergencies.	The HI-STORM UMAX closure lid can be removed to gain access to the multi-purpose canister. All HI-TRAC transfer cask models have removable bottom and top lids.	Chapter 9 Chapter 9
viii. A confinement barrier that acceptably protects the spent fuel cladding during storage.	The peak temperature of the fuel cladding at design basis heat duty of each MPC has been demonstrated to be maintained below the limits specified in ISG-11 Revision 3. The confinement barriers consist of highly ductile stainless steel alloys. The multi-purpose canister is housed in the VVM, built from a steel structure whose materials are selected and examined to maintain protection against brittle fracture under off-normal ambient (cold) temperatures (minimum of -40ºF).	Section 4.4 Subsection 3.1
ix. The structures are compatible with the appropriate monitoring systems.	The HI-STORM UMAX VVM with openings that are designed to prevent radiation streaming while enabling complete access to temperature monitoring probes, if used.	Section 1.5

Table 3.7.1

NUREG-1536 COMPLIANCE MATRIX FOR 10CFR72.120 AND 10CFR72.122 REQUIREMENTS

Item		Compliance	Notes
x.	Structural designs are compatible with ready retrievability of fuel.	The VVMs are designed to withstand accident loads without suffering permanent deformations of their structures that would prevent retrievability of the MPC by normal means. It is demonstrated by analysis that there is no physical interference between the MPC and the enveloping HI-STORM VVM under all postulated Design Basis loads.	Section 3.4
xi.	Adequate heat removal without active cooling systems.	Thermal analyses presented in Chapter 4 show that the HI-STORM UMAX System will remove the decay heat generated from the stored spent fuel by strictly passive means and maintain the system temperature within prescribed limits.	Section 4.4
xii.	Storage of spent fuel for a minimum of 20 years.	The service life of the storage modules is engineered to be in excess of 60 years.	Subsection 3.4.7
xiii.	Conspicuous and durable marking.	The exterior envelope of the VVM is marked in a conspicuous manner as required by 10CFR 72.236(k).	N/A
ix.	Compatibility with removal of the stored fuel from the site, transportation, and ultimate disposal by the U.S. Department of Energy.	The MPCs used in the HI-STORM UMAX System are intended to serve as a transportable waste package.	N/A

3.8 REFERENCES

[3.1.1] ANSYS 11.0, ANSYS Inc., 2007.

[3.1.2] H. Boyer, Atlas of Stress Strain Curves, ASM International, 1987.

[3.1.3] Doug Ammerman and Gordon Bjorkman, "Strain-Based Acceptance Criteria for Section III of the ASME Boiler and Pressure Vessel Code", Proceedings of the 15th International Symposium on the Packaging and Transportation of Radioactive Materials, PATRAM 2007, October 21-26, 2007, Miami, Florida, USA.

[3.1.4] ASME Boiler and Pressure Vessel Code, Section III, Division 1, Subsection NB, 2010.

[3.1.5] ASME Boiler and Pressure Vessel Code, Section III, Division 1, Subsection NF - Supports, 2010.

[3.1.6] "Construction of True-Stress-True-Strain Curves for LS-DYNA Simulations," Holtec Proprietary Position Paper DS-307, Revision 2.*

[3.1.7] HI-2114807, Thermal-Hydraulic Evaluation of HI-STORM UMAX, Revision 0, 2011 (Holtec Proprietary).

[3.3.1] ASTM C39, Standard Test Method for Compressive Strength of Cylindrical Concrete.

[3.3.2] ASME Boiler & Pressure Vessel Code, Section II, Part D, 2010.

[3.3.3] B.H.Hough, Basic Soils Engineerings, Second Edition.

[3.3.4] Holtec Position Paper, DS-338, "A Methodology to Compute the Equivalent Elastic Properties of the Subgrade Continuum to Incorporate the Effect of Long-Term Settlement," A.I. Soler and C. Bullard (2010) (Holtec Proprietary)

[3.4.1] HI-2125228, Structural Calculation Package for the HI-STORM UMAX System, Latest Revision, 2012 (Holtec Proprietary).

[3.4.2] American National Standards Institute, ANSI B30.3

* Supporting document submitted with the HI-STORM FW License Application (Docket 72-1032).

[3.4.3] ANSI N14.6-1993, "American National Standard for Special Lifting Devices for Shipping Containers Weighing 10,000 Pounds (4500 kg) or More for Nuclear Materials", June 1993.

[3.4.4] SHAKE2000, A Computer Program for the 1-D Analysis of Geotechnical Earthquake Engineering Problems, G.A. Ordonez, Dec. 2000.

[3.4.5] LS-DYNA, Version 971, Livermore Software.

[3.4.6] ACI-318(2005), Building Code Requirements for Structural Concrete (ACI 318-05) and Commentary (ACI 318R-05), American Concrete Institute, 2005.

[3.4.7] Holtec Position Paper, DS-338, "A Methodology to Compute the Equivalent Elastic Properties of the Subgrade Continuum to Incorporate the Effect of Long-Term Settlement," A.I. Soler and C. Bullard (2010) (Holtec Proprietary).

[3.4.8] ACI-349 (2001), Code Requirements for Nuclear Safety Related Concrete Structures (ACI 349-01) and Commentary (ACI 349R-01), Appendix C, American Concrete Institute, 2001.

[3.4.9] ACI 360R-06, Design of Slabs on Grade, American Concrete Institute, 2006.

[3.4.10] Sandia National Laboratory Contractor Report SAND97-2426, Penetration Equations, C.Y. Young, Applied Research Associates, Inc., Albuquerque NM 87110.

[3.4.11] SAE Technical Paper 2000-01-0627, Development and Validation of High Fidelity Vehicle Crash Simulation Models, S.W. Kirkpatrick, Applied Research Associates, Inc.

[3.4.12] Topical Report – Design of Structures for Missile Impact", BC-TOP-9A, Rev. 2, Bechtel Corporation, 9/74

[3.4.13] 10CFR71, Waste Confidence Decision Review, USNRC, September 11, 1990.

[3.4.14] ASCE 4-98, Seismic Analysis of Safety-Related Nuclear Structures and Commentary, American Society of Civil Engineers, 2000.

[3.4.15] ASCE/SEI 43-05, Seismic Design Criteria for Structures, Systems, and Components in Nuclear Facilities, American Society of Civil Engineers, 2005.

[3.4.16] ASLB Hearings, Private Fuel Storage, LLC, Docket # 72-22-ISFSI, ASLBP 97-732-02-ISFSI, February 2005.

[3.4.17] HI-2125239, Structural Analysis of HI-STORM UMAX ISFSI Structures, Latest

Revision, 2012 (Holtec Proprietary).

[3.6.1] Validation Manual for SHAKE 2000; Holtec Report Number # HI-2104792 (2010)

CHAPTER 4: THERMAL EVALUATION

4.0 OVERVIEW

HI-STORM UMAX is an underground vertical ventilated module (VVM) with openings for air ingress and egress and internal air flow passages for ventilation cooling of loaded MPC. The licensing drawing package for the HI-STORM UMAX is provided in Section 1.5. Thermal design requirements are presented in Section 2.5. The analyses summarized in this chapter focus on the governing canisters out of the population of MPCs listed in Table 1.2.1. This chapter, however, supports the certification of only MPC-37 and MPC-89 at this time. The analyses reported for smaller canisters are for reference purposes only.

Section 1.2 provides a summary description of the HI-STORM UMAX system. The MPC types considered for evaluating storage in HI-STORM UMAX VVMs are listed in Table 1.2.1. In this chapter, compliance of HI-STORM UMAX system's thermal performance to 10CFR72 requirements for storage at an ISFSI using 3-D thermal simulation models is established. The analyses consider passive rejection of decay heat from the stored SNF assemblies to the environment under normal, off-normal, and accident conditions of storage. In particular, the thermal margins of safety for long-term storage of both moderate burnup (up to 45,000 MWD/MTU) and high burnup spent nuclear fuel (greater than 45,000 MWD/MTU) in the HI-STORM UMAX system are quantified. The HI-STORM UMAX deploys MPCs and HI-TRAC transfer casks that have been previously certified in HI-STORM FW FSAR and CoC (USNRC Docket 72-1032). Safe thermal performance during fuel loading, unloading and on-site transfer operations, collectively referred to as "short-term operations" utilizing the HI-TRAC VW transfer cask for MPC-37 and MPC-89 is also evaluated. The cases of normal, off-normal and accident conditions of storage, enumerated in Chapter 2 are also evaluated for the MPC designs in Table 1.2.1 to establish an acceptable safety case for their long term storage in the HI-STORM UMAX VVMs.

The thermal evaluation of HI-STORM UMAX follows the guidelines of NUREG-1536 [4.0.1] and ISG-11 [4.0.2]. These guidelines provide specific limits on the permissible maximum cladding temperature in the stored commercial spent fuel (CSF)* and other Confinement Boundary components, and on the maximum permissible pressure in the confinement space under certain operating scenarios. Specifically, the requirements are:

1. The fuel cladding temperature must meet the temperature limit appropriate to its burnup level and condition of storage / handling set forth in NUREG-1536 [4.0.1] and ISG-11 [4.0.2].

2. The maximum internal pressure of the MPC should remain within its design pressures for normal, off-normal, and accident conditions set forth in Table 2.3.5.

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* Defined as nuclear fuel that is used to produce energy in a commercial nuclear reactor (See Glossary).

3. The temperatures of the cask materials shall remain below their allowable limits set forth in Table 2.3.7 under all scenarios.

Section 2.5 of this FSAR contains a listing of all thermal analysis cases that warrant analysis. Section 2.1 in Chapter 2 provides the Design Basis heat loads (maximum permissible heat loads in storage cells). As demonstrated in this chapter, the HI-STORM UMAX system is designed to comply with all of the thermal criteria defined in Chapter 2.

Sections 4.1 through 4.3 describe thermal analyses and input data that are common to all conditions of storage, handling and on-site transfer operations. All required thermal analyses to evaluate normal conditions of storage in a HI-STORM UMAX storage module are described in Section 4.4. The thermal performance of the system is also evaluated under sustained wind conditions in Section 4.4. Thermal analyses to evaluate on-site transfer in a HI-TRAC transfer cask are considered in Section 4.5 and the evaluations to establish compliance under off-normal and accident conditions are summarized in Section 4.6.

To facilitate convenient access to the referenced material, the latest edition of the HI-STORM FW FSAR has been placed in this docket and a list of "FW" FSAR sections germane to this chapter is provided in a tabular form. The HI-STORM FW FSAR will be maintained in a configuration controlled status in this docket as a mandatory supplement to this FSAR. Table 4.0.1 provides a listing of the material adopted in this chapter by reference to the HI-STORM FW FSAR.

The safety analyses summarized in this chapter demonstrate acceptable margins to the allowable limits under all design basis loading conditions and operational modes. Minor changes to the design parameters that inevitably occur during the product's life cycle which are treated within the purview of 10CFR72.48 and are ascertained to have an insignificant effect on the computed safety factors may not prompt a formal reanalysis and revision of the results and associated data in the tables of this chapter unless the cumulative effect of all such unquantified changes on the reduction of any of the computed safety margins cannot be deemed to be insignificant. For purposes of this determination, an insignificant loss of safety margin with reference to an acceptance criterion is defined as the estimated reduction that is no more than one order of magnitude below the available margin reported in the FSAR. To ensure rigorous configuration control, the information in the Licensing drawings in Section 1.5 should be treated as the authoritative source for numerical analysis at all times. Reliance on the input data and associated results in this chapter for additional mathematical computations may not be appropriate as they serve the sole purpose of establishing safety compliance in accordance with the acceptance criteria set down in Chapter 2 and in this chapter.

Table 4.0.1

HI-STORM FW FSAR MATERIAL GERMANE TO THE EVALUATIONS IN THIS FSAR *

Location of UMAX FSAR	Subject of the reference	Location in HI-STORM FW FSAR, Revision 3
Sub-Section 4.4.1	Description of MPC thermal models	Sub-Section 4.4.1
Sub-Section 4.4.1	Fuel region effective thermal properties	Paragraph 4.4.1.1
Sub-Section 4.4.1	Flow resistance through fuel assemblies	Paragraph 4.4.1.2
Paragraph 4.6.1.4	FHD Malfunction	Paragraph 4.6.1.4
Paragraph 4.6.2.1	Fire accident events	Paragraph 4.6.2.1
Paragraph 4.6.2.6	HI-TRAC VW jacket water loss accident	Paragraph 4.6.2.2

* For convenience of reference, the specific revision of the HI-STORM FW FSAR that is referenced in the safety analysis herein is placed in this docket. Updated versions of the HI-STORM FW FSAR shall be placed in this docket as necessary so as to ensure that the safety analyses on the "UMAX" docket (72-1040) remain aligned with the material referenced in the HI-STORM FW FSAR.

4.1 DESIGN BASIS HEAT LOAD AND GOVERNING MPC FOR THERMAL ANALYSIS

As stated in Section 2.0, the HI-STORM UMAX system has been designed to store all previously certified MPCs under NRC dockets No. 72-1014 and 72-1032. Therefore it is necessary to identify the governing MPC from the list in Table 1.2.1 along with its governing heat load. The governing MPC and its heat load combination are defined as the one that yields the maximum PCT. For this purpose, it is helpful to consider two fundamental facts relevant to the thermal performance of VVMs; namely,

a. Smaller the size of the flow annulus, smaller the air flow rate and a correspondingly lesser extent of heat extraction from the MPC surface and hence greater the PCT in the stored fuel.
b. Larger the heat load, greater the PCT. It must be noted that both the total heat load and per assembly heat load distribution must be considered.

Because the inside diameter of HI-STORM UMAX VVM is fixed, it is readily deduced that the larger diameter MPCs previously licensed in the HI-STORM FW docket [4.1.2] will have a smaller axial flow annulus compared to those in the HI-STORM 100 docket [4.1.1]. The DBHL is also substantially greater in the HI-STORM FW system compared to the HI-STORM 100 system. The input data for all MPCs previously licensed in HI-STORM FW docket, i.e. MPC-37 with short, standard and long fuels under various heat load charts (see Table 2.1.8) and MPC-89 (see Table 2.1.9) are summarized in Table 4.1.1. The maximum peak cladding temperature results are calculated and reported in Table 4.1.2. The results demonstrate that MPC-37 with the short fuel under Heat Load Chart 1 produces the highest PCT and it is therefore designated as the governing thermal scenario.

It is noted that the MPCs in Docket # 72-1014 are smaller in diameter and are certified to lower DBHLs. Nevertheless, the hottest MPC from the HI-STORM 100 docket also warrants analysis because the stainless steel fuel baskets in it are less conductive than their counterparts in the HI-STORM FW docket (that contain METAMIC-HT fuel baskets). As noted in the HI-STORM 100 FSAR, MPC-32 under X=3* regionalized loading scenario (see Section 2.1.10) results in the highest PCT. Therefore it is the governing case among all MPCs in the HI-STORM 100 docket [4.1.1] and therefore is also selected as a candidate MPC type (with regionalized loading scenario X=3) for thermal analysis in HI-STORM UMAX. The case of X=0.5 (see Sub-Section 2.1.10), which is the other limit of regionalization ratio, is also included for completeness.

All the thermal scenarios for MPC-37 and MPC-89 are defined in Table 4.1.1. The limiting thermal scenario for an MPC-32 is defined based on the information in its existing host docket and summarized in Table 4.1.1. The thermal problem is then analyzed using the solution methodology approved in [4.1.1] and [4.1.2]. The temperatures presented in Tables 4.1.2 show

* X is defined in the HI-STORM 100 FSAR [4.1.1] as the ratio of maximum permissible storage per cell heat load in the inner and outer regions.

that the ISG-11 Rev 3 limits are satisfied with comfortable margins.

The maximum peak cladding temperature results for MPC-37, MPC-89 and MPC-32, reported in Table 4.1.2, show that MPC-37 with short fuel under Heat Load Chart 1 produces the highest PCT. Therefore, MPC-37 with short fuel under Heat Load Chart 1 fulfills the criterion for designation as the governing thermal configuration. This governing thermal configuration is used to perform the thermal safety analyses that are reported in the subsequent sections in this chapter.

Table 4.1.1

CANDIDATE GOVERNING CANISTERS & HEAT LOAD CASES STORAGE IN THE HI-STORM UMAX SYSTEM

Item	**MPC-37 (Docket 72-1032)**	**MPC-89 (Docket 72-1032)**	**MPC-32**[Note 1] **(Docket 72-1014)**
MPC Decay Heat	Table 2.1.8[Note 2]	Table 2.1.9[Note 2]	30.17 kW @X=3 36.9 kW @X=0.5
MPC Minimum Backfill Pressure	Table 4.4.6	Table 4.4.6	Table 4.4.6
MPC Maximum Backfill Pressure	Table 4.4.6	Table 4.4.6	Table 4.4.6
Normal Ambient Temperature	Table 2.3.6	Table 2.3.6	Table 2.3.6
Temperature of the Subgrade Supporting the Support Foundation Pad	Table 2.3.6	Table 2.3.6	Table 2.3.6
MPC External Diameter[Note 3]	75-3/4”	75-3/4”	68-3/8”
Divider Shell Inner Diameter[Note 3]	86”	86”	86”
Insulation Thickness	2-3/4”	2-3/4”	2-3/4”
Inner Diameter of Container Shell[Note 3]	100”	100”	100”

Note 1: MPC-32 bounds MPC-68, 68M and MPC-24 as noted in HI-STORM 100 Docket 72-1014.

Note 2: All heat load charts for MPC-37 and MPC-89 provided in Section 2.1.9 are evaluated.

Note 3: UMAX System may have an MPC with increased shell thickness with corresponding increased MPC external diameter and UMAX Divider and Container Shell inner diameters. Such systems are equivalent to or bounded by the existing thermal analysis for an MPC listed in this table.

Table 4.1.2

PEAK CLADDING TEMPERATURE RESULTS FOR LONG-TERM NORMAL STORAGE FOR MPC-32, MPC-37 AND MPC-89 IN HI-STORM UMAX SYSTEM**

MPC Types		**Fuel Cladding Temperature ^{o}C (^{o}F)**
MPC-37 (Heat Load Chart 1)	Short Fuel	**367 (693)***
	Standard Fuel	357 (675)
	Long Fuel	351 (664)
MPC-37 (Heat Load Chart 2)	Short Fuel	363 (685)
	Standard Fuel	355 (671)
	Long Fuel	346 (655)
MPC-37 (Heat Load Chart 3)	Short Fuel	359 (678)
	Standard Fuel	364 (687)
	Long Fuel	353 (667)
MPC-89		357 (675)
MPC-32 (X=3)		366 (691)

* Based on the results in this table, MPC-37 with short fuel under Heat Load Chart 1 is selected as the governing thermal configuration and is used to perform all the licensing basis calculations for HI-STORM UMAX System.
** The PCT results documented in this table are for normal storage conditions under quiescent (no wind) conditions.

4.2 SUMMARY OF THERMAL PROPERTIES OF MATERIALS

The thermo-physical properties listed in the tables in this section are identical to those used in the HI-STORM FW FSAR [4.1.2] and in the HI-STORM 100 FSAR [4.1.1]. Materials present in the MPCs listed in Table 1.2.1 include Alloy X*, Metamic-HT, aluminum, and helium. Materials present in the HI-STORM UMAX storage overpack include carbon steel, stainless steel, insulation, and concrete (in the Closure Lid). In Table 4.2.1, a summary of references used to obtain cask material properties for performing all thermal analyses is presented.

Tables 4.2.2 and 4.2.3 provide numerical thermal conductivity data of materials at several representative temperatures. Table 4.2.4 provides the data on emissivity.

In Table 4.2.5, the heat capacity and density of the MPC, overpack and CSF materials are presented. These properties are used in performing transient (i.e., 32 hours inlet duct block and hypothetical fire accident condition) analyses. The temperature-dependent values of the viscosities of helium and air are provided in Table 4.2.6.

To minimize heating of the air in the intake down flow passage, the overpack divider shell is thermally insulated. The density and specific heat of the insulation material are conservatively assumed to be the same as air. The upper bound insulation conductivity is specified in Table 4.2.7.

The properties listed in the tables in this section are consistent across all previously established HI-STORM dockets.

* Alloy X is defined in [4.1.1] to designate a group of stainless steel alloys permitted for use in the HI-STORM MPCs. In this chapter the terms Alloy X and stainless steel are used interchangeably.

Table 4.2.1

SUMMARY OF HI-STORM UMAX SYSTEM MATERIALS
THERMAL PROPERTY REFERENCES

Material	**Emissivity**	**Conductivity**	**Density**	**Heat Capacity**
Helium	N/A	Handbook [4.2.2]	Ideal Gas Law	Handbook [4.2.2]
Air	N/A	Handbook [4.2.2]	Ideal Gas Law	Handbook [4.2.2]
Zircaloy	[4.2.3], [4.2.13], [4.2.14], [4.2.7]	NUREG [4.2.13]	Rust [4.2.4]	Rust [4.2.4]
UO_2	Note 1	NUREG [4.2.13]	Rust [4.2.4]	Rust [4.2.4]
Stainless Steel (machined forgings)[Note 2]	Kern [4.2.5]	ASME [4.2.8]	Marks' [4.2.1]	Marks' [4.2.1]
Stainless Steel Plates[Note 3]	ORNL [4.2.11], [4.2.12]	ASME [4.2.8]	Marks' [4.2.1]	Marks' [4.2.1]
Carbon Steel	Kern [4.2.5]	ASME [4.2.8]	Marks' [4.2.1]	Marks' [4.2.1]
Concrete	Incropera [4.2.9]	Marks' [4.2.1]	Chapter 2	Handbook [4.2.2]
Water	Note 1	ASME [4.2.10]	ASME [4.2.10]	ASME [4.2.10]
Metamic-HT	Test Data [4.2.6]	Test Data [4.2.6]	Test Data [4.2.6]	Test Data [4.2.6]
Insulation	Table 4.2.7	Table 4.2.7	Table 4.2.7	Table 4.2.7
Aluminum	Test Data [4.2.6]	ASM [4.2.15]	ASM [4.2.15]	ASM [4.2.15]

Note 1: Emissivity not reported as radiation heat dissipation from these surfaces is conservatively neglected.

Note 2: Used in the MPC lid. May be used as an optional material for CEC inlet plenum air-intake flange.

Note 3: Used in the MPC shell and baseplate. May be used as an optional material for some CEC components, as allowed by the licensing drawing in Section 1.5.

Table 4.2.2

SUMMARY OF HI-STORM UMAX SYSTEM MATERIALS
THERMAL CONDUCTIVITY DATA

Material	**At 200°F (Btu/ft-hr-°F)**	**At 450°F (Btu/ft-hr-°F)**	**At 700°F (Btu/ft-hr-°F)**	**At 1000°F (Btu/ft-hr-°F)**
Helium	0.0976	0.1289	0.1575	0.1890
Air*	0.0173	0.0225	0.0272	0.0336
Alloy X	8.4	9.8	11.0	12.4
Carbon Steel	24.4	23.9	22.4	20.0
Concrete**	1.05	1.05	1.05	1.05
Water	0.392	0.368	N/A	N/A
Metamic-HT	**[PROPRIETARY INFORMATION WITHHELD IN ACCORDANCE WITH 10 CFR 2.390]**			
Aluminum**	69.3	69.3	69.3	69.3

* At lower temperatures, Air conductivity is between 0.0139 Btu/ft-hr-°F at 32°F and 0.0176 Btu/ft-hr-°F at 212°F.

** Conservatively assumed to be constant for the entire range of temperatures.

Table 4.2.3*

SUMMARY OF FUEL ELEMENT COMPONENTS
THERMAL CONDUCTIVITY DATA

Zircaloy Cladding		**Fuel (UO_2)**	
Temperature (°F)	Conductivity (Btu/ft-hr-°F)	Temperature (°F)	Conductivity (Btu/ft-hr-°F)
392	8.28	100	3.48
572	8.76	448	3.48
752	9.60	570	3.24
932	10.44	793	2.28

* See Table 4.2.1 for cited references.

Table 4.2.4

SUMMARY OF MATERIALS SURFACE EMISSIVITY DATA*

Material	Emissivity
Zircaloy	0.80
Painted surfaces	0.85
Stainless steel (machined forgings)	0.36
Stainless Steel Plates	0.587**
Carbon Steel	0.66
Concrete	0.88**
Metamic-HT***	**[PROPRIETARY INFORMATION WITHHELD IN ACCORDANCE WITH 10 CFR 2.390]**
Aluminum***	**[PROPRIETARY INFORMATION WITHHELD IN ACCORDANCE WITH 10 CFR 2.390]**

* See Table 4.2.1 for cited references.

** Lower bound value from the cited references in Table 4.2.1.

*** **[PROPRIETARY INFORMATION WITHHELD IN ACCORDANCE WITH 10 CFR 2.390]**

Table 4.2.5

DENSITY AND HEAT CAPACITY PROPERTIES SUMMARY*

Material	Density (lbm/ft^3)	Heat Capacity (Btu/lbm-^{o}F)
Helium	(Ideal Gas Law)	1.24
Air	(Ideal Gas Law)	0.24
Zircaloy	409	0.0728
Fuel (UO_2)	684	0.056
Carbon steel	489	0.1
Stainless steel	501	0.12
Concrete	140**	0.156
Water	62.4	0.999
Metamic-HT	**[PROPRIETARY INFORMATION WITHHELD IN ACCORDANCE WITH 10 CFR 2.390]**	
Aluminum	177.3	0.207

* See Table 4.2.1 for cited references.

** Conservatively understated value used for the Closure Lid concrete only.

Table 4.2.6

GASES VISCOSITY* VARIATION WITH TEMPERATURE

Temperature (^oF)	**Helium Viscosity (Micropoise)**	**Temperature (^oF)**	**Air Viscosity (Micropoise)**
167.4	220.5	32.0	172.0
200.3	228.2	70.5	182.4
297.4	250.6	260.3	229.4
346.9	261.8	338.4	246.3
463.0	288.7	567.1	293.0
537.8	299.8	701.6	316.7
737.6	338.8	1078.2	377.6
921.2	373.0	-	-
1126.4	409.3	-	-

* See Table 4.2.1 for applicable reference.

Table 4.2.7

THERMAL PROPERTIES OF INSULATION

Density, kg/m^3	1.2 Note 2
Heat Capacity, J/kg-K	1004 Note 2
Thermal Conductivity, W/m-K	0.072 Note 1
Surface Emissivity	0.3 Note 2

Note 1: Important to Safety property (see Section 4.2).
Note 2: NITS properties.

4.3 SPECIFICATIONS FOR COMPONENTS

Permissible temperatures for the HI-STORM UMAX system materials and components designated as “Important to Safety” (i.e., required to be maintained within their safe operating temperature ranges to ensure their intended function) are summarized in Table 2.3.7. Long-term integrity of SNF is ensured by the HI-STORM UMAX system thermal evaluation which demonstrates that fuel cladding temperatures are maintained below design basis limits.

Compliance to 10CFR72 requires, in part, identification and evaluation of short-term, off-normal and hypothetical accident conditions. The inherent mechanical characteristics of cask materials and components ensure that no significant functional degradation is possible due to exposure to short-term temperature excursions outside the normal long-term temperature limits. Fuel temperature limits specified in ISG-11 [4.0.2] are compiled in Table 2.3.7 for evaluation of cladding integrity under normal, short term operations, off-normal, and accident conditions.

4.4 THERMAL EVALUATION FOR NORMAL CONDITIONS OF STORAGE

The HI-STORM UMAX System thermal evaluation is performed in accordance with the guidelines of NUREG-1536 [4.0.1] and ISG-11 [4.0.2]. Table 1.2.1 lists the canister types and their host dockets where their thermal qualifications are performed. The candidate canisters and the heat load charts are identified and summarized in Table 4.1.1. These candidate thermal configurations, as mentioned in Section 4.1, are analyzed using the FLUENT model described below to identify the "governing thermal configuration", defined as the one that yields the highest PCT.

4.4.1 FLUENT Thermal Model

The thermal analysis model for the HI-STORM UMAX system utilizes the MPC models for MPC-37 and MPC-89 described in reference [4.1.2] and that for MPC-32 described in [4.1.1]. The effective properties of the fuel storage cells used in the thermal analysis of MPC-37 and MPC-89 in the HI-STORM UMAX System are conservatively 10% lower than those reported in the HI-STORM FW FSAR [4.1.2]. A geometrically accurate 3D thermal model of the HI-STORM UMAX VVM is constructed in the manner of HI-STORM 100U in Docket number 72-1014 for analysis. The VVM closure lid, inlet and outlet duct, the inner and outer annulus, the U-turn and the air plenum above the MPC are explicitly modeled.

The airflow through the cooling passages of the VVM is modeled **[PROPRIETARY INFORMATION WITHHELD IN ACCORDANCE WITH 10 CFR 2.390]** recommended in the Holtec-proprietary benchmarking report [4.4.1]. This is the *same* modeling approach as used in docket numbers 72-1014 and 72-1032. The underside of the VVM Support Foundation Pad is assumed to be supported on a subgrade at an isothermal surface temperature (see Table 4.1.1). The VVM thermal models are constructed using the same modeling platform used for aboveground analysis (FLUENT version 6.3).

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4.4.2 Grid Sensitivity Studies

To insure fully converged CFD results, a grid sensitivity study was performed. **[**

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A number of grids were generated to study the effect of mesh refinement on the component temperatures. All sensitivity analyses were carried out for the case of MPC-37 with short fuel under Heat Load Chart 1, which is determined to be the governing thermal configuration in Section 4.1. Table 4.4.1 gives a brief summary of the different sets of grids evaluated and PCT results.

Table 4.4.1 shows that Mesh 3 and Mesh 5 report essentially the same results. Therefore it can be concluded that Mesh 3 is reasonably converged [4.4.7]. To provide further assurance of convergence, the uncertainty of CFD result was evaluated in accordance with the ASME V&V 20-2009 for control of numerical accuracy [4.4.2]. The Grid Convergence Index (GCI), which is a measure of the solution uncertainty, is computed to be 1.068% for Meshes 1, 3 and 5. This provides further assurance of grid convergence.

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It is noted that Mesh 3 (moderate mesh size) yields a slightly higher PCT than Mesh 5 (largest mesh size). Considering the computational cost and available PCT margin, the Mesh configuration 3 was conservatively used in all thermal analyses reported herein.

4.4.3 Test Model

The HI-STORM UMAX thermal analysis is performed on the FLUENT [4.4.3] Computational Fluid Dynamics (CFD) program. To ensure a high degree of confidence in the HI-STORM UMAX thermal evaluations, the FLUENT code has been benchmarked using data from tests conducted with casks loaded with irradiated SNF ([4.4.4], [4.4.6]). The benchmark work is archived in QA validated Holtec reports ([4.4.1], [4.4.5]). These evaluations show that the FLUENT solutions are conservative in all cases. In view of these considerations, additional experimental verification of the thermal design is not necessary. Furthermore, FLUENT has been relied to secured certification in all Holtec International Part 71 and Part 72 dockets.

4.4.4 Maximum and Minimum Temperatures

i. Maximum Temperatures

A comprehensive set of thermal analyses of all candidate “thermal configurations” (meaning the combination of canister type, regionalized loading pattern and fuel type that may produce highest fuel cladding temperature) were performed using the FLUENT model described in Section 4.4.1 to quantify their thermal margins under long term storage conditions. Thermal analyses of the following cases are performed:

A. MPC-37 with short, standard and long fuel and MPC-89 under conditions described in Table 4.1.1*.
B. Docket # 72-1014: MPC-32 under two extreme regionalized loading scenarios (X=0.5 and X=3) described in Table 4.1.1.

The maximum spatial values of the computed temperatures of the fuel cladding, the fuel basket material, the divider shell, the closure lid concrete, the MPC lid, the MPC shell and the average air outlet for all cases are summarized in Table 4.4.2. It can be seen that the governing case is that of MPC-37 with short fuel under heat load chart 1.

The following conclusions are reached from the solution data:

a. The PCT is below the temperature limit set forth in NUREG-1536 [4.0.1] and ISG-11 [4.0.2].
b. The maximum temperatures of all MPC and VVM constituent parts are below their respective limits set down in Table 2.3.7.

It is therefore concluded that the HI-STORM UMAX system provides a thermally acceptable storage environment for the eligible MPCs listed in Table 1.2.1 and that the governing thermal

* Design basis heat load scenario bounds sub-design basis and threshold heat load scenarios defined in Tables 4.4.3, 4.4.4 and 4.4.5 (See Section 4.4.5).

configuration corresponds to the case of MPC-37 with short fuel (Heat Load Chart 1): All other thermal configurations yield a lower PCT result.

ii. Minimum Temperatures

In Table 2.3.6 of this report, the minimum ambient temperature condition for the HI-STORM UMAX storage overpack and MPC is specified to be -40 deg. F. If, conservatively, a zero decay heat load with no solar input is applied to the stored fuel assemblies, then every component of the system at steady state would be at a temperature of -40 deg. F. Low service temperature (-40^{o}F) evaluation of the HI-STORM UMAX is provided in Chapter 8. All HI-STORM UMAX storage overpack and MPC materials of construction will satisfactorily perform their intended function in the storage mode under this minimum temperature condition.

4.4.5 Maximum Internal Pressure in the MPC

The MPC is initially filled with dry helium after fuel loading and drying prior to installing the MPC closure ring. In the MPC host docket (72-1014), the different helium backfill pressure specifications are allowed according to the different CoC amendments. For each type of MPC in Docket 72-1014, the specification of helium backfill pressure for HI-STORM UMAX System envelop all allowed specifications defined in the MPC host docket. The helium backfill pressure for the MPCs authorized to be stored in HI-STORM UMAX are presented in Table 4.4.6.

To provide additional helium backfill range for less than design basis heat load, the following Sub-Design-Basis (SDB) heat load scenarios are allowed for MPC-37 and MPC-89:

(i) MPC-37 under 90% of Heat Load Chart 1 (see Table 4.4.3)
(ii) MPC-89 under 90% of Design Heat Load (see Table 4.4.4)
(iii) MPC-37 under threshold heat load (see Table 4.4.5)
(iv) MPC-89 under threshold heat load (see Table 4.4.5)

The storage cell and MPC heat load limits under the 90% of Design Heat Load scenarios mentioned above are obtained by multiplying 0.9 to the individual storage cell of design heat load, and reported in Tables 4.4.3 and 4.4.4. The storage cell and MPC heat load limits under the threshold heat load scenario in the HI-STORM UMAX are provided in Table 4.4.5.

The helium backfill pressure limits supporting these scenarios are summarized in Table 4.4.6. These backfill limits maybe optionally adopted by a cask user if the decay heats of the loaded fuel assemblies meet the SDB decay heat limits stipulated above.

The heat load of SDB scenarios (i) thru (iv) are bounded by the Design Heat Load scenarios. However, since the helium backfill specifications for SDB scenarios are different from the design basis heat load scenarios, confirmatory analysis under UMAX storage are required. Considering that MPC-37 loaded with short fuel yields the highest temperatures, explicit calculations under threshold heat load and 90% of design basis heat load are performed. The

principal results are reported in Table 4.4.10. The results comply with the temperature and pressure limit Tables 2.3.7 and 2.3.5. The results support the conclusion that threshold heat load and SDB scenarios are bounded by Chart 1 Design Heat Load scenario.

During normal storage, the gas temperature within the MPC rises to its maximum operating basis temperature. The gas pressure inside the MPC will also increase with rising temperature. The pressure rise is determined using the ideal gas law. The MPC gas pressure is also subject to substantial pressure rise under hypothetical rupture of fuel rods and large gas inventory non-fuel hardware (PWR BPRAs).

The MPC maximum gas pressure is computed for a postulated release of fission product gases from fuel rods into this free space. For these scenarios, the amounts of each of the release gas constituents in the MPC cavity are summed and the resulting total pressures determined from the ideal gas law. A concomitant effect of rod ruptures is the increased pressure and molecular weight of the cavity gases with enhanced rate of heat dissipation by internal helium convection and lower cavity temperatures. As these effects are substantial under large rod ruptures the 100% rod rupture accident is evaluated with due credit for increased heat dissipation under increased pressure and molecular weight of the cavity gases. Based on fission gases release fractions (NUREG 1536 criteria [4.0.1]), rods' net free volume and initial fill gas pressure, the maximum gas pressures with 1% (normal), 10% (off-normal) and 100% (accident condition) rod rupture are given in Table 4.4.7 for all thermal scenarios analyzed in Section 4.4.4. The maximum calculated gas pressures reported in Table 4.4.7 are all below the MPC internal design pressures for normal, off-normal and accident conditions specified in Table 2.3.5.

4.4.6 Engineered Clearances to Eliminate Thermal Interferences

Thermal stress in a structural component is the resultant sum of two factors, namely: (i) restraint of free end expansion and (ii) non-uniform temperature distribution. To minimize thermal stresses in load bearing members, the HI-STORM UMAX system is engineered with adequate gaps to permit free thermal expansion of the fuel basket and MPC in axial and radial directions. In this subsection, differential thermal expansion calculations are performed for the governing thermal configuration, i.e. MPC-37 with short fuel under Heat Load Chart 1, to demonstrate that engineered gaps in the HI-STORM UMAX System are adequate to accommodate thermal expansion of the fuel basket and MPC.

The following gaps are evaluated:

a. Fuel Basket-to-MPC Radial Gap
b. Fuel Basket-to-MPC Axial Gap
c. MPC-to-Divider Shell Radial Growth
d. MPC-to-VVM Closure Lid Axial Growth

The FLUENT thermal model provides the 3-D temperature field in the HI-STORM UMAX system from which the changes in the above gaps are directly computed. Table 4.4.8 provides the nominal gaps and their corresponding value during long-term storage conditions. Significant margins against restraint to free-end expansion are indicated by the data in Table 4.4.8.

4.4.7 Effect of Elevation

The reduced ambient pressure at site elevations significantly above the sea level will act to reduce the ventilation air mass flow, resulting in a net elevation of the peak cladding temperature. However, the ambient temperature (i.e., temperature of the feed air entering the overpack) also drops with the increase in elevation. Because the peak cladding temperature also depends on the feed air temperature (the effect is one-for-one within a small range, i.e., 1°F drop in the feed air temperature results in ~1°F drop in the peak cladding temperature), the adverse ambient pressure effect of increased elevation is partially offset by the ambient air temperature decrease. The table below illustrates the variation of air pressure and corresponding ambient temperature as a function of elevation.

Elevation (ft)	Pressure (psia)	Ambient Temperature Reduction versus Sea Level
Sea Level (0)	14.70	0°F
2000	13.66	7.1°F
4000	12.69	14.3°F

A survey of the elevation of nuclear plants in the U.S. shows that nuclear plants are situated near about sea level or elevated slightly (~1000 ft). The effect of the elevation on peak fuel cladding temperatures is evaluated by performing calculations for a HI-STORM UMAX system situated at an elevation of 1500 feet. At this elevation the ambient temperature would decrease by approximately 5°F (See Table above). The peak cladding temperatures are calculated under the reduced ambient temperature and pressure at 1500 feet elevation for the MPC-37 with the short fuel under heat load chart 1. The results are presented in Table 4.4.9.

These results show that the PCT, including the effects of site elevation, continues to be well below the regulatory cladding temperature limit of 752°F. In light of the above evaluation, it is not necessary to place ISFSI elevation constraints for HI-STORM UMAX deployment at elevations up to 1500 feet. If, however, an ISFSI is sited at an elevation greater than 1500 feet, the effect of altitude on the PCT shall be quantified as part of the 10 CFR 72.212 evaluation for the site using the site ambient conditions.

4.4.8 Burnup Effects on Thermal Performance of HI-STORM UMAX System

[PROPRIETARY INFORMATION WITHHELD IN ACCORDANCE WITH 10 CFR 2.390]

[PROPRIETARY INFORMATION WITHHELD IN ACCORDANCE WITH 10 CFR 2.390] Thermal conductivities of fresh UO_2 were used to determine the effective thermal properties of the fuel storage cells. However, it is known that the thermal conductivity of the fuel pellet (UO_2) will reduce with increasing burn-ups. The progressive buildup of fission products with increasing fuel burnup in the fuel pellets progressively reduces its thermal conductivity [4.4.9]. The effect of reduction in the fuel pellet thermal conductivity with fuel burnup and therefore, on the effective planar thermal conductivities of the rodded region is evaluated in [4.4.8]. **[**

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For heat transfer in the axial direction, the thermal conductivity of the fuel storage cells is determined via an area weighted mean of the fuel cladding and helium conductivities. Axial conduction heat transfer in the fuel pellets is ignored per the guidance provided in NUREG-1536 [4.0.1]. Therefore, a change in thermal conductivity of UO_2 will have no effect on the axial (or longitudinal) thermal conductivity of the fuel storage cells.

The effective properties of the fuel storage cells used in the thermal analysis of MPC-37 and MPC-89 in the HI-STORM UMAX System are conservatively 10% lower than that reported in HI-STORM FW FSAR [4.1.2]. Since the decrease in effective thermal conductivity of fuel due to high burnup effects is bounded by the values used in the licensing basis analysis, no further thermal evaluations of HI-STORM UMAX System are warranted.

4.4.9 Evaluation of Sustained Wind

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4.4.10 Evaluation of System Performance for Normal Conditions of Storage

The HI-STORM UMAX System thermal analysis is based on a detailed 3-D heat transfer model that conservatively accounts for all modes of heat transfer in the MPC and overpack. The thermal model incorporates conservative assumptions that render the computed temperature results for long-term storage to be conservative. The computed temperatures in “UMAX” under the governing thermal scenarios show that in each case:

a. The peak cladding temperature is below the ISG-11 Rev 3 limit.
b. The temperatures of structural members in the VVM, which are made of either carbon steel or stainless steel, is well below their allowable values set down in the Chapter 2 (presented in Table 2.3.7).
c. The temperature of shielding concrete mass and insulation (both non-structural members) are also well within their stipulated limits set forth in Table 2.3.7

The modest metal temperatures reached in “UMAX” insure that the components of the system will not suffer long term degradation from elevated temperature effects such as creep, alloy

phase transformation, recrystallization of the materials' grain structure, and the like. Therefore, safety of long term storage from the thermal standpoint is assured.

4.4.11 Effects of Optional Shielding

To reduce the occupational dose during loading operation, an optional stainless steel ring can be implemented. The location and specification of shielding ring are given in the drawing package listed in Section 1.5. To study the effect of this optional shielding ring, MPC-37 with short fuel under heat load chart 1, is re-evaluated for normal long-term storage condition under quiescent conditions. The PCT, maximum component temperatures of MPC and HI-STORM UMAX VVM are reported in Table 4.4.18. The MPC cavity pressure is also reported in Table 4.4.12. These results comply with the temperature and pressure limits reported in Tables 2.3.7 and 2.3.5, respectively. Comparing Table 4.4.18 with the MPC-37 in Table 4.4.2 shows that the impact of this shielding ring is insignificant.

Table 4.4.1

NORMAL LONG-TERM STORAGE TEMPERATURES FOR LIMITING MPC-37 WITH SHORT FUEL (HEAT LOAD CHART 1) USING DIFFERENT MESHES

Mesh No	Total Cell Number	PCT °C (°F)	Permissible PCT Limit °C (°F)	PCT Margin °C (°F)
1	932,307	372 (702)	400 (752)	28 (50)
3*	2,075,968	367 (693)	400 (752)	33 (59)
5	4,724,915	364 (687)	400 (752)	36 (65)

* Mesh 3 is reasonably converged and is adopted for all licensing basis calculation.

Table 4.4.2

NORMAL LONG-TERM STORAGE TEMPERATURES FOR CANDIDATE CANISTERS UNDER DESIGN BASIS HEAT LOADS

Component	**MPC-37 with Short Fuel (Heat Load Chart 1)** **°C (°F)**	**MPC-89 Design Basis Heat Load** **°C (°F)**	**MPC-32 under Heat Load Pattern X=0.5** **°C (°F)**	**MPC-32 under Heat Load Pattern X=3** **°C (°F)**
Fuel Cladding	367 (693)	357 (675)	363 (685)	366 (691)
MPC Basket	353 (667)	348 (658)	361 (682)	364 (687)
Basket Periphery	286 (547)	287 (549)	302 (576)	274 (525)
Aluminum Basket Shims	263 (505)	266 (511)	-	-
MPC Shell	232 (450)	242 (468)	230 (446)	217 (423)
MPC Lid*	238 (460)	247 (477)	246 (475)	241 (466)
Divider Shell	169 (336)	176 (349)	150 (302)	134 (273)
Containment Shell	55 (131)	55 (131)	53 (127)	52 (126)
Closure Lid Concrete†	104 (219)	107 (225)	97 (207)	89 (192)
Insulation	169 (336)	176 (349)	150 (302)	134 (273)
Average Air Outlet‡	75 (167)	74 (165)	74 (165)	69 (156)

* Maximum section average temperature reported.

† Maximum section average temperature reported.

‡ Section average temperature on the cross section area of outlet duct below the outlet vent screen reported. Reported herein for the option of temperature measurement surveillance of outlet duct air temperature as set forth in the Technical Specifications. The bounding air outlet temperature of 160°F corresponding to MPC-37 with long fuel is specified in the Technical Specifications.

Table 4.4.3

HI-STORM UMAX MPC-37 SUB-DESIGN BASIS HEAT LOAD DATA

Sub-Design Heat Chart Type	Type of Fuel	Permissible Heat Load per Storage Cell	Heat Load (kW)
90% of Chart 1	Short and Standard Fuel	Figure 2.1.19	38.08[Note 1]
	Long Fuel	Figure 2.1.20	40.21[Note 2]

Note 1: Thermal evaluations were performed for the heat load tabulated herein. However, the CoC restricts the aggregate heat load to 34.28 kW.

Note 2: Thermal evaluations were performed for the heat load tabulated herein. However, the CoC restricts the aggregate heat load to 36.19 kW.

Table 4.4.4

HI-STORM UMAX MPC-89 SUB-DESIGN BASIS HEAT LOAD DATA

Sub-Design Heat Chart Type	**Permissible Heat Load per Storage Cell**	**Heat Load (kW)**
90% of Design Basis Heat Load	Figure 2.1.23	40.8[Note1]
Note 1: Thermal evaluations were performed for the heat load tabulated herein. However, the CoC restricts the aggregate heat load to 36.72 kW.		

Table 4.4.5

HI-STORM UMAX THRESHOLD HEAT LOAD DATA[Note 1]

MPC Type	Permissible Heat Load per Storage Cell	Heat Load (kW)
MPC-37	Figure 2.1.21	33.46
MPC-89	Figure 2.1.24	34.75
Note 1: Vacuum drying operations are allowed for MPCs containing high burnup fuel assemblies only up to the threshold heat load tabulated herein.		

Table 4.4.6

MPC HELIUM BACKFILL SPECIFICATIONS*

MPC Type	Helium Backfill Pressure Option	Specification†, psig
MPC-37	1	≥ 41.0 and ≤ 44.2
	2	≥ 41.0 and ≤ 44.5
	3	≥ 39.0 and ≤ 46.0
MPC-89	1	≥ 42.0 and ≤ 45.2
	2	≥ 39.0 and ≤ 46.0
MPC-32		≥ 43.7 and ≤ 48.5
MPC-24/24E/24EF		≥ 42.5 and ≤ 48.5
MPC-68/68F/68FF/ 68M		≥ 43.5 and ≤ 48.5
Note: The permissible aggregate heat loads and per cell storage limits MPC-37 and MPC-89 for each helium backfill pressure option are provided in Table 2.1.8.		

* The helium backfill specification for each MPC in Docket 72-1014 envelops the helium backfill specification defined in its host docket.

† Specification at a reference temperature of 21.1°C (70°F).

Table 4.4.7

SUMMARY OF MPC INTERNAL PRESSURES UNDER THE LIMITING THERMAL SCENARIOS FOR LONG-TERM STORAGE (QUIESCENT CONDITIONS)*

Condition	**MPC-37 with Short Fuel (Heat Load Chart 1) (psig)**	**MPC-89 Design Basis Heat Load (psig)**	**MPC-32 under Heat Load Pattern X=0.5 (psig)**	**MPC-32 under Heat Load Pattern X=3 (psig)**
Normal:				
intact rods	94.0	94.6	96.6	89.9
1% rods rupture	95.1	95.2	97.6	90.8
Off-Normal (10% rods rupture)	104.9	100.3	106.3	99.0
Accident (100% rods rupture)	189.6	151.4	193.7	181.1

* Per NUREG-1536, pressure analyses with ruptured fuel rods (including BPRA rods for PWR fuel) is performed with release of 100% of the ruptured fuel rod fill gas and 30% of the significant radioactive gaseous fission products.

Table 4.4.8

SUMMARY OF HI-STORM UMAX DIFFERENTIAL THERMAL EXPANSIONS FOR LIMITING MPC-37 WITH SHORT FUEL (HEAT LOAD CHART 1)

Gap Description	**Cold Gap U (in)**	**Differential Expansion δ_i (in)**	**Is Free Expansion Criterion Satisfied (i.e., $U > \delta_i$)**
Fuel Basket-to-MPC Radial Gap	0.125	0.121	Yes
Fuel Basket-to-MPC Axial Gap	1.5	0.443	Yes
MPC-to-Divide Shell Radial Gap	0.25	0.147	Yes
MPC-to-Closure Lid Axial Gap	23.5	0.504	Yes

Table 4.4.9

NORMAL LONG-TERM STORAGE TEMPERATURES FOR LIMITING MPC-37 WITH SHORT FUEL (UNDER HEAT LOAD CHART 1) AT AN ELEVATED SITE

Component	**Temperature ^{o}C (^{o}F)**
Fuel Cladding	368 (694)
MPC Basket	354 (669)
Basket Periphery	287 (549)
Aluminum Basket Shims	265 (509)
MPC Shell	234 (453)
MPC Lid*	242 (468)
Divider Shell	171 (340)
Containment Shell	53 (127)
Closure Lid Concrete†	104 (219)
Insulation	171 (340)
Average Air Outlet‡	75 (167)

* Maximum section average temperature reported.
† Maximum section average temperature reported.
‡ Section average temperature on the cross section area of outlet duct below the outlet vent screen reported. Reported herein for the option of temperature measurement surveillance of outlet duct air temperature as set forth in the Technical Specifications

Table 4.4.10

NORMAL LONG-TERM STORAGE TEMPERATURES AND PRESSURE FOR MPC-37 WITH SHORT FUEL UNDER SUBDESIGN HEAT LOADS

Component	90% of Chart 1 Temperature °C (°F)	Threshold Heat Load Temperature °C (°F)
Fuel Cladding	346 (655)	304 (579)
MPC Basket	332 (630)	291 (556)
Basket Periphery	267 (513)	242 (468)
Aluminum Basket Shims	246 (475)	224 (435)
MPC Shell	217 (423)	199 (390)
MPC Lid*	223 (433)	199 (390)
Divider Shell	157 (315)	142 (288)
Containment Shell	54 (129)	52 (126)
Closure Lid Concrete†	98 (208)	90 (194)
Insulation	157 (315)	143 (289)
MPC Cavity Pressure (psig)		
Normal Condition – No Rods Rupture	94.0	89.3

* Maximum section average temperature reported.
† Maximum section average temperature reported.

Table 4.4.11
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Table 4.4.12

EFFECT OF WIND ON PEAK CLADDING TEMPERATURE - A SINGLE HI-STORM UMAX SYSTEM SIMULATED

Wind Speed	**Fuel Cladding Temperature ^{o}C (^{o}F)**
0 MPH	367 (693)*
2 MPH	371 (700)
5 MPH	377 (711)
7 MPH	378 (712)
8 MPH	379 (714)
9 MPH	379 (714)
10 MPH	377 (711)
12 MPH	378 (712)
15 MPH	376 (709)
17 MPH	373 (703)
20 MPH	369 (696)

* Reproduced from Table 4.4.2 for the quarter symmetric model.

Table 4.4.13

EFFECT OF WIND ON AIR INLET TEMPERATURES OF HI-STORM UMAX SYSTEMS STORED IN AN ISFSI ARRAY

	Air Inlet Temperature Note 1, °F				
	0 MPH Note 2	2 MPH	5 MPH	7 MPH	10 MPH
Module 1 Note 3	82	84	84	84	84
Module 2	82	83	85	86	86
Module 3	82	85	86	86	88
Module 4	82	90	86	89	90
Module 5	82	**93**	88	89	91
Module 6	82	91	89	89	92
Module 7	82	91	**93**	90	92
Module 8	82	90	90	87	89

Note 1: Air inlet temperature reported in this table is the mass-average temperature calculated in the cross-section surface of inlet pipe immediate below the inlet vent screen.

Note 2: Reference from the quarter symmetric model described in Section 4.4.4.

Note 3: The series of module is numbered in the sequence as the wind direction, i.e. Module 1 is the front module facing the wind inlet.

Table 4.4.14

EFFECT OF WIND ON AIR MASS FLOW RATE THROUGH HI-STORM UMAX SYSTEM STORED IN AN ISFSI ARRAY

	Air Mass Flow Rate*, kg/s				
	0 MPH	2 MPH	5 MPH	7 MPH	10 MPH
Module 1†	0.801	0.751	0.646	0.622	0.614
Module 2	0.801	0.780	0.790	0.762	0.732
Module 3	0.801	0.771	0.758	0.756	0.742
Module 4	0.801	0.764	0.752	0.744	0.746
Module 5	0.801	0.748	0.744	0.758	0.748
Module 6	0.801	0.698	0.746	0.742	0.734
Module 7	0.801	0.726	0.746	0.742	0.746
Module 8	0.801	0.734	0.704	0.750	0.736

* Air inlet mass flow rate reported in this table is calculated in the cross-section surface of inlet pipe immediate below the inlet vent screen.

† The series of module is numbered in the sequence as the wind direction, i.e. Module 1 is the front module facing the wind inlet.

Table 4.4.15

MAXIMUM HI-STORM UMAX STEADY STATE STORAGE TEMPERATURES UNDER WIND CONDITION

Component	**Temperature °C (°F)**
Fuel Cladding	386 (727)
MPC Basket	372 (702)
Basket Periphery	302 (576)
Aluminum Basket Shims	279 (534)
MPC Shell	247 (477)
MPC Lid*	250 (482)
Divider Shell	188 (370)
Containment Shell	62 (144)
Closure Lid Concrete†	120 (248)
Insulation	188 (370)

Table 4.4.16

EFFECT OF WIND DIRECTION ON MAXIMUM HI-STORM UMAX STEADY STATE STORAGE TEMPERATURES

Item	**PCT, °C (°F)**
Parallel Flow	378 (712)
Oblique Flow	381 (718)

* Maximum section average temperature reported.
† Maximum section average temperature reported

Table 4.4.17

SUMMARY OF MPC INTERNAL PRESSURES FOR THE LIMITING THERMAL SCENARIO (MPC-37 WITH SHORT FUEL UNDER HEAT LOAD CHART 1) UNDER WORST SCENARIO WIND CONDITION

Condition*	**Gauge Pressure (psig)**
Normal: intact rods 1% rods rupture	 98.6 99.7
Off-Normal (10% rods rupture)	110.0
Accident (100% rods rupture)	198.9

* Per NUREG-1536, pressure analyses with ruptured fuel rods (including BPRA rods for PWR fuel) is performed with release of 100% of the ruptured fuel rod fill gas and 30% of the significant radioactive gaseous fission products.

Table 4.4.18

NORMAL LONG-TERM STORAGE TEMPERATURES AND PRESSURE
WITH OPTIONAL SHIELDING RING
FOR MPC-37 WITH SHORT FUEL UNDER HEAT LOAD CHART 1

Component	**Temperature °C (°F)**
Fuel Cladding	368 (694)
MPC Basket	354 (669)
Basket Periphery	285 (545)
Aluminum Basket Shims	263 (505)
MPC Shell	232 (450)
MPC Lid*	229 (444)
Divider Shell	170 (338)
Containment Shell	54 (129)
Closure Lid Concrete†	98 (208)
Insulation	170 (338)
MPC Cavity Pressure (psig)	
Normal Condition – No Rods Rupture	94.1

* Maximum section average temperature reported.
† Maximum section average temperature reported.

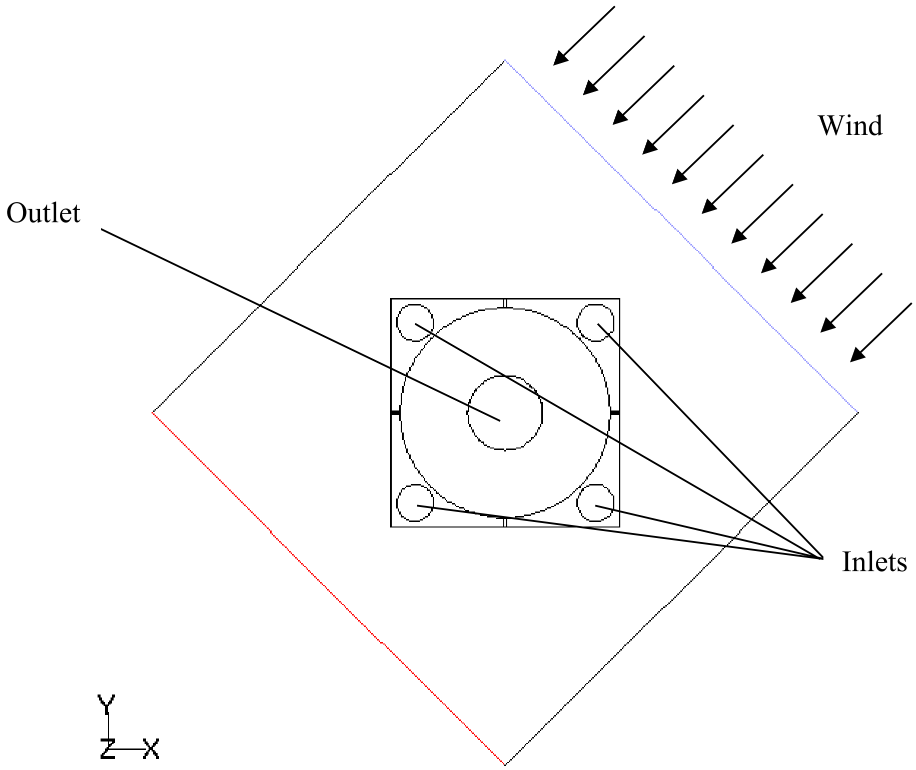

Figure 4.4.1: Schematic of the Analysis Condition - HI-STORM UMAX Full Model (Oblique Flow)

4.5 SHORT-TERM OPERATIONS

Short-Term Operations are those activities that are required to load and package the fuel in a waste package (i.e., a multi- purpose canister) and to place it in long term storage at an ISFSI. In the regulatory literature, all activities that originate in the plant's spent fuel pool and culminate in the MPC emplaced on the ISFSI pad in a passive storage mode or vice-verse from ISFSI to pool under the unlikely scenario requiring fuel unloading are collectively referred to as Short Term Operations. These include operations that occur inside the part 50 facility such as dewatering, drying, and backfilling the canister, establishing the confinement boundary by welding the lid and the Closure Ring and those that occur outside of it, namely transporting the canister to the ISFSI and transferring the MPC to the storage module. All of the short term operations have one common feature: They all involve the transfer cask. In fact, the beginning and end of Short Term Operations can be identified by the moment the fuel enters the MPC inside HI-TRAC to the time when the MPC is delivered to the storage module. The qualification of Short Term Operations, therefore, is integral to the certification of the transfer cask. This FSAR envisages using MPCs and their associated transfer casks certified in the HI-STORM FW FSAR [4.1.2]. The transfer operation of MPC-37 and MPC-89 in HI-TRAV VW is evaluated in the following.

The operational steps that occur during MPC transfer operations include the Mating Device installed and the drawer closed. As explained below this operational step must be performed in an expeditious manner to avoid excessive heating of the MPC and fuel (See Section 9.2). As the Mating Device in the closed drawer position blocks air flow it must be opened to establish air cooling. In the event of equipment malfunction that results in the blockage of air flow, corrective actions must occur within the time limits of the 100% blocked duct accident condition evaluated in Section 4.6. During MPC transfer operation from HI-TRAC to the UMAX (or reverse operation to support fuel unloading), air flow through the VVM cavity will be completely stopped under a complete closure of the Mating Device drawer. This scenario is same as the all inlet duct blockage condition analyzed in Section 4.6.2.3. Therefore, the temperature rise of fuel with time under this event, as shown in Figure 4.6.1 is applicable to evaluation of the closed drawer event. The maximum allowable time duration to ensure temperature limits of limiting fuel (high burnup fuel) remain within short term limits is computed as 4 hours. The time limit computed herein is for an example case of design basis heat load and MPCs containing one or more high burnup fuel assemblies. Additional site-specific analysis may be performed to compute applicable time limits under less than design-basis loadings.

4.5.1 Thermally Limiting Evolutions During Short-Term Operations

Except for MPC-to-UMAX transfer operations evaluated in Section 4.5 herein thermally limiting evolutions under short-term operations defined in the HI-STORM FW FSAR Section 4.5.1 [4.1.2] are incorporated by reference.

4.5.2 HI-TRAC VW Thermal Model

The HI-TRAC VW transfer cask is used to load and unload the HI-STORM UMAX concrete

storage overpack, including onsite transport of the MPCs from the loading facility to an ISFSI pad. The HI-TRAC VW 3-D thermal model is exactly the same as that used for evaluation of the thermal state of a loaded canister during all short-term operations defined in the HI-STORM FW FSAR Section 4.5.2 [4.1.2] except for the following enhancements:

1. The emissivity of fuel basket and aluminum basket shims is based on the values tabulated in Table 4.2.4.

The HI-TRAC VW thermal model is evaluated under the limiting scenario of fuel storage in the minimum height MPC-37 (See Section 4.4.1.5) and limiting Chart 1 heat load (See Section 4.4.4) specified in Chapter 2. Results of on-site transfer analyses are provided in Subsection 4.5.4.

4.5.3 Maximum Time Limit During Wet Transfer Operations

Time limits evaluated under wet transfer operations in the HI-STORM FW FSAR Section 4.5.3 [4.1.2] are incorporated by reference.

4.5.4 Analysis of Limiting Thermal States During Short-Term Operations

4.5.4.1 Vacuum Drying

Vacuum drying evaluations for MPC-37 is exactly the same as that in HI-STORM FW FSAR Section 4.5.4.1 except for the following major additional changes:

1. Vacuum drying of MPCs containing high burnup fuel assemblies is permitted up to threshold heat loads defined in Table 4.4.5.

2. A flattened axial heat generation distribution for PWR fuel assemblies shown in Table 2.1.5 is replaced by a center biased power distribution defined in Table 4.5.1. The center biased power distribution is conservative and results in higher peak cladding temperatures.

The results from the analysis performed for this condition, using the 3-D Fluent model, are provided in Table 4.5.2. The results provide a sufficiently comfortable cushion against the 400^{o}C (752^{o}F) limit.

Vacuum drying evaluations for MPC-89 in the HI-STORM FW FSAR Section 4.5.4.1 [4.1.2] are incorporated by reference.

4.5.4.2 Forced Helium Dehydration

Forced Helium Dehydration in the HI-STORM FW FSAR Section 4.5.4.2 [4.1.2] are incorporated by reference.

4.5.4.3 Normal On-site Transfer

An MPC-37 situated inside a HI-TRAC VW is evaluated under the limiting thermal scenario i.e. short fuel under heat load Chart 1. This scenario is analyzed using the same 3D FLUENT model of the HI-TRAC VW transfer cask in HI-STORM FW FSAR with due recognition of the changes discussed in Section 4.5.2 above.

While the duration of onsite transport is generally short to preclude the MPC and HI-TRAC VW from reaching a steady-state, a conservative approach is adopted herein by assuming steady state maximum temperatures are reached. The principal objectives of the HI-TRAC VW analyses are to demonstrate:

i) Cladding integrity
ii) Confinement integrity
iii) Neutron shield integrity

The appropriate criteria are provided in Tables 2.3.5 (pressure limits) and 2.3.7 (temperature limits).

The results of thermal analyses tabulated in Table 4.5.3 show that the cladding temperatures are below the ISG-11 temperature limits of High and Moderate Burnup Fuel (Table 2.3.7). Actual margins during HI-TRAC VW operations will be much larger due to the many conservative assumptions incorporated in the analysis.

The predicted temperatures and MPC cavity pressure in Tables 4.5.3 are still bounded by the respective counterpart temperatures reported for normal on-site transfer in Section 4.5 of HI-STORM FW FSAR [4.1.2]. Therefore, the accident conditions pertaining to HI-TRAC discussed in HI-STORM FW FSAR remain bounding as well and no additional analysis is warranted in this FSAR.

4.5.5 Cask Cooldown and Reflood During Fuel Unloading Operation

Cask cooldown and reflood evaluation in the HI-STORM FW FSAR Section 4.5.5 [4.1.2] is incorporated by reference.

4.5.6 Maximum Internal Pressure

After fuel loading and vacuum drying, but prior to installing the MPC closure ring, the MPC is initially filled with helium. During handling and on-site transfer operations in the HI-TRAC VW transfer cask, the gas temperature will correspond to the thermal conditions within the MPC analyzed in Section 4.5.4.3. Based on this analysis the MPC internal pressure is computed under the assumption of maximum helium backfill specified in Table 4.4.6 and confirmed to comply

with the short term operations pressure limit in Table 2.3.5. The results are tabulated in Table 4.5.3.

TABLE 4.5.1

NORMALIZED PWR DISTRIBUTION BASED ON A CENTER BIASED BURNUP PROFILE

Interval	**Axial Distance From Bottom of Active Fuel (% of Active Fuel Length)**	**Normalized Distribution**
1	0% to 4-1/6%	0.78
2	4-1/6% to 8-1/3%	0.78
3	8-1/3% to 16-2/3%	0.78
4	16-2/3% to 33-1/3%	1.10
5	33-1/3% to 50%	1.20
6	50% to 66-2/3%	1.15
7	66-2/3% to 83-1/3%	1.07
8	83-1/3% to 91-2/3%	0.7
9	91-2/3% to 95-5/6%	0.70
10	95-5/6% to 100%	0.70

TABLE 4.5.2

MAXIMUM COMPONENT TEMPEARTURES DURING VACUUM DRYING OPERATIONS OF MPC-37 AT THRESHOLD HEAT LOAD

Component	**Temperature °C (°F)**
Fuel Cladding	383 (721)
MPC Basket	367 (693)
Basket Periphery	291 (550)
Aluminum Basket Shims	233 (450)
MPC Shell	142 (288)
MPC Lid*	100 (212)

* Maximum section average temperature is reported.

TABLE 4.5.3

HI-TRAC VW TRANSFER CASK STEADY STATE MAXIMUM TEMPERATURES FOR MPC-37 WITH SHORT FUEL UNDER HEAT LOAD CHART 1

Component	**Temperature °C (°F)**
Fuel Cladding	378 (712)
MPC Basket	364 (687)
Basket Periphery	291 (556)
Aluminum Basket Shims	265 (509)
MPC Shell	242 (468)
MPC Lid*	233 (451)
HI-TRAC Inner Shell	136 (277)
HI-TRAC Top Flange	131 (268)
HI-TRAC Bottom Flange	142 (288)
HI-TRAC Radial Lead Gamma Shield	135 (275)
Water Jacket outer Shell	128 (262)
Water Jacket Bulk Water	127 (261)
MPC Cavity Pressure (psig)	
No Rods Rupture	97.0

* Maximum section average temperature is reported

4.6 OFF-NORMAL AND ACCIDENT EVENTS

The safety evaluation of off-normal and accident conditions described in Section 2.5 is presented in this section. Thermal analysis of the HI-STORM UMAX System is performed for the "governing thermal configuration", i.e. MPC-37 with short fuel under heat load Chart 1, identified by the analysis in Section 4.1.

4.6.1 Off-Normal Events

4.6.1.1 Off-Normal Environmental Temperature

To evaluate the effect of off-normal weather conditions, an off-normal ambient temperature (Table 2.3.6) is postulated to persist for a sufficient duration to allow the HI-STORM UMAX system to reach steady state conditions. Because of the large mass of the HI-STORM UMAX system, with its corresponding large thermal inertia and the limited duration for the off-normal temperatures that arise in real life, this assumption is conservative. Starting from a baseline condition evaluated in Section 4.4 (normal ambient temperature and limiting fuel storage configuration) the temperatures of the HI-STORM UMAX system are conservatively assumed to be elevated by the difference between the off-normal and normal ambient temperatures. The HI-STORM UMAX extreme ambient temperatures computed in this manner are reported in Table 4.6.1. The co-incident MPC pressure is also computed (Table 4.6.5) and compared with the off-normal design pressure (Table 2.3.5), which shows a positive safety margin. The results are confirmed to be below the corresponding limits in Chapter 2.

4.6.1.2 Partial Blockage of Air Inlets

The HI-STORM UMAX system is designed with debris screens installed on the inlet and outlet openings. These screens ensure the air passages are protected from entry and blockage by foreign objects. However, as required by the design criteria presented in Chapter 2, it is postulated that the HI-STORM UMAX air inlet vents are 50% blocked. The resulting decrease in flow area increases the flow resistance of the inlet ducts. The effect of the increased flow resistance on fuel temperature is analyzed assuming that steady state conditions have been reached for the "governing thermal configuration" established by a series of thermal analyses in Section 4.4 in the foregoing. The computed temperatures and pressures are reported in Tables 4.6.1 and 4.6.5 respectively. The results are confirmed to be below the allowable limits for both internal pressure and temperature limits presented in Tables 2.3.5 and 2.3.7 respectively.

4.6.1.3 Off-Normal Pressure

This event is defined as a combination of (a) maximum helium backfill pressure permitted, (b) 10% fuel rods rupture, (c) governing thermal configuration, and (d) normal ambient temperature defined in Table 2.3.6. The principal objective of the analysis is to demonstrate that the MPC off-normal design pressure (Table 2.3.5) is not exceeded. Table 4.4.7 provides the computed pressures for the off-normal event as defined above which show that all applicable pressure

limits are met with positive margins.

4.6.1.4 FHD Malfunction

FHD malfunction evaluated in the HI-STORM FW FSAR Section 4.6.1 [4.1.2] is incorporated by reference.

4.6.2 Accident Events

4.6.2.1 Fire Accident

(a) HI-STORM UMAX Fire
The Design Basis Fire event for HI-STORM UMAX, described in Section 2.5, is identical to that of HI-STORM FW[4.1.2] and HI-STORM 100 [4.1.1]. The fire evaluation for limiting MPC-37 with short fuel stored in HI-STORM UMAX is bounded by the analysis reported in the HI-STORM FW FSAR [4.1.2], due to the following facts:

- The initial PCT and component temperatures of MPC stored in HI-STORM UMAX system are lower than that of the same MPC in the HI-STORM FW system.
- HI-STORM UMAX system has much lesser surface directly exposed to fire than that of above-ground system.

Consequently, the conclusion that PCT and components' temperatures and MPC pressure are below temperature and pressure limits for the Design Basis Fire event drawn in HI-STORM FW FSAR [4.1.2] will remain valid for the HI-STORM UMAX system.

(b) HI-TRAC VW Fire
The thermal evaluation of normal on-site transfer in HI-TRAC VW discussed in Sub-Section 4.5.4 is bounded by the normal on-site transfer evaluation of HI-TRAC VW in the HI-STORM FW FSAR Section 4.5.4 [4.1.2]. Therefore, HI-TRAC VW fire evaluated in the HI-STORM FW FSAR Section 4.6.2 [4.1.2] is incorporated by reference.

4.6.2.2 Extreme Environmental Temperatures

To evaluate the effect of extreme weather conditions, an extreme ambient temperature (Table 2.3.6) is postulated to persist for a sufficient duration to allow the HI-STORM UMAX system to reach steady state conditions. Because of the large mass of the HI-STORM UMAX system, with its corresponding large thermal inertia and the limited duration for the extreme temperatures that arise in real life, this assumption is conservative. Starting from a baseline condition evaluated in Section 4.4 (normal ambient temperature and limiting fuel storage configuration) the temperatures of the HI-STORM UMAX system are conservatively assumed to be elevated by the

difference between the extreme and normal ambient temperatures. The HI-STORM UMAX extreme ambient temperatures computed in this manner are reported in Table 4.6.6. The co-incident MPC pressure is also computed (Table 4.6.10) and compared with the accident design pressure (Table 2.3.5), which shows a positive safety margin. The results are confirmed to be below the corresponding limits in Chapter 2.

4.6.2.3 100% Blockage of Air Inlets

This event is defined as a postulated complete blockage of all inlet ducts for a specified duration. The immediate consequence of a complete blockage of the air inlets is that the normal circulation of air for cooling the MPC is interrupted. A small amount of heat will continue to be removed by localized air circulation patterns in the VVM annulus and outlet ducts, and the MPC will continue to dissipate heat to the relatively cooler subgrade. As the temperatures of the MPC and its contents rise, the rate of heat rejection will increase correspondingly. Nevertheless, under this condition, the temperatures of the storage system including the MPC and the stored fuel assemblies will rise monotonically as a function of time.

As a result of the considerable inertia of the storage overpack, a significant temperature rise is possible if the inlets are substantially blocked for extended durations. This accident condition is, however, a short duration event that is identified and corrected through scheduled periodic surveillance. Nevertheless, this event is conservatively analyzed assuming a substantial duration of blockage. The inlet ducts in the HI-STORM UMAX thermal model are assumed to have become impervious to air flow for 100% air inlet blockage. Using this model, a transient thermal solution of the HI-STORM UMAX system starting from normal storage conditions is obtained. The results of the 32 hours blocked ducts transient analysis are presented in Table 4.6.7 and compared against the accident temperature limits (Table 2.3.7). The history of PCT increasing during the early time stage is plotted in Figure 4.6.1 to provide guideline for fuel loading operation. The co-incident MPC pressure (Table 4.6.10) is also computed and compared with the accident design pressure (Table 2.3.5). All computed results are found to remain well below their respective limits under the postulated 32 hours blockage duration.

4.6.2.4 Burial Under Debris

Burial of the HI-STORM UMAX system under debris is not a credible accident. During storage at the ISFSI there are no structures that loom over the casks whose collapse could completely bury the casks in debris. Minimum regulatory distances from the ISFSI to the nearest ISFSI security fence precludes the close proximity of substantial amount of vegetation. Thus, even though there is no credible mechanism for the HI-STORM UMAX system to become completely buried under debris, the scenario of complete burial under debris is analyzed herein to quantify the system's resistance to such an event.

Thus, burial under debris is a postulated accident scenario that is analyzed in this FSAR to determine the length of time that is available to the plant's emergency response organization to remedy the condition. The generic burial-under-debris analysis is performed under the following

assumptions:

a. The ISFSI is assumed to be populated with the limiting MPC type containing the most adverse thermal loading, i.e. MPC-37 with short fuel under heat load chart 1.

b. The burial medium serves as a perfect insulation blocking off any conductive or convective means for heat rejection to the environment.

c. The burial is assumed to affect all VVMs at the ISFSI. Thus no heat transmission path to the adjacent subgrade is available.

d. Heat rejection to the under-grade is also assumed to be (non-mechanistically) lost.

Under the above scenario, the contents of the HI-STORM UMAX system will undergo a transient heat up under adiabatic conditions. The minimum available time ($\Delta\tau$) for the fuel cladding to reach the accident limit depends on the following: (i) thermal inertia of the cask, (ii) the cask initial conditions, (iii) the spent nuclear fuel decay heat generation and (iv) the margin between the initial cladding temperature and the accident temperature limit. To obtain a *lower bound* on $\Delta\tau$, the HI-STORM UMAX VVM's thermal inertia (item i) is understated, the cask initial temperature (item ii) is maximized, decay heat overstated (item iii) and the cladding temperature margin (item iv) is understated. A set of conservatively postulated input parameters for items (i) through (iv) are summarized in Table 4.6.8. Using these parameters $\Delta\tau$ is computed as follows:

$$\Delta\tau = \frac{m \times c_p \times \Delta T}{Q}$$

where:

$\Delta\tau$ = minimum available burial time (hr)
m = Mass of HI-STORM UMAX System (lb)
c_p = Specific heat capacity (Btu/lb-^{o}F)
ΔT = Permissible temperature rise (^{o}F)
Q = Decay heat load (Btu/hr)

Substituting the parameters in Table 4.6.8, the minimum available burial time is computed as 22.7 hours. This is an example calculation for the short fuel. The same methodology can be adopted to estimate the time for other MPC types and fuel assemblies. The co-incident MPC pressure (see Table 4.6.10) is also computed and compared with the accident design pressure (Table 2.3.5). These results indicate that HI-STORM UMAX has a substantial thermal heat sink capacity to withstand a complete burial-under-debris event.

The above simplified computation may be used to compute the permissible burial-under-debris time for a VVM cavity corresponding to its actual content conditions. The available time to

restore full ventilation in a loaded VVM is computed above for guiding the Emergency Response plan for an ISFSI for which a burial-under-debris event can be credibly postulated.

4.6.2.5 Flood

[

PROPRIETARY INFORMATION WITHHELD IN ACCORDANCE WITH 10 CFR 2.390

]

[
PROPRIETARY INFORMATION WITHHELD IN ACCORDANCE WITH 10 CFR 2.390
]

4.6.2.6 Jacket Water Loss

The thermal evaluation of normal on-site transfer in HI-TRAC VW discussed in Sub-Section 4.5.4 is bounded by the normal on-site transfer evaluation of HI-TRAC VW in the HI-STORM FW FSAR Section 4.5.4 [4.1.2]. Therefore, HI-TRAC VW jacket water loss accident evaluated in the HI-STORM FW FSAR Section 4.6.2 [4.1.2] is incorporated by reference.

Table 4.6.1

OFF-NORMAL EVENTS - MAXIMUM TEMPERATURES

Component	**Off-Normal Ambient Temperature Condition °C (°F)**	**Partial Inlet Ducts Blockage Condition °C (°F)**
Fuel Cladding	378 (713)	369 (696)
MPC Basket	364 (687)	356 (673)
Basket Periphery	297 (568)	294 (561)
Basket Shims	274 (525)	274 (525)
MPC Shell	243 (470)	240 (464)
MPC Lid*	249 (480)	246 (475)
Divider Shell	180 (356)	177 (351)
Containment Shell	66 (151)	56 (133)
Closure Lid Concrete†	115 (239)	109 (228)
Insulation	180 (356)	177 (351)

* Maximum section average temperature reported.
† Maximum section average temperature reported.

Table 4.6.5

OFF-NORMAL CONDITION MAXIMUM MPC PRESSURES FOR LIMITING MPC-37 WITH SHORT FUEL UNDER HEAT LOAD CHART 1

Condition	**Gauge Pressure (psig)**
Off-Normal Ambient Temperature	96.2
Partial Blockage of Inlet Ducts	95.4

Table 4.6.6

EXTREME ENVIRONMENTAL CONDITION
MAXIMUM HI-STORM UMAX TEMPERATURES

Component	**Temperature** ^{o}**C (**^{o}**F)**
Fuel Cladding	392 (738)
MPC Basket	378 (712)
Basket Periphery	311 (592)
Aluminum Basket Shims	288 (550)
MPC Shell	257 (495)
MPC Lid*	263 (505)
Divider Shell	194 (381)
Containment Shell	80 (176)
Closure Lid Concrete†	129 (264)
Insulation	194 (381)

* Maximum section average temperature reported.
† Maximum section average temperature reported.

Table 4.6.7

MAXIMUM TEPERATURES REACHED AFTER 32 HOURS OF COMPLETE DUCT BLOCKAGE(GOVERNING THERMAL CONFIGURATION CASE)

Component	**Final Condition ^oC (^oF)**
Fuel Cladding	518 (964)
MPC Basket	502 (936)
Basket Periphery	438 (820)
Aluminum Basket Shims	417 (783)
MPC Shell	389 (732)
MPC Lid*	333 (631)
Divider Shell	368 (694)
Containment Shell	246 (475)
Closure Lid Concrete†	217 (423)
Insulation	367 (693)

* Maximum section average temperature reported.
† Maximum section average temperature reported.

Table 4.6.8

SUMMARY OF INPUTS FOR BURIAL UNDER DEBRIS ANALYSIS FOR SHORT FUEL

Thermal Inertia Inputs: M (Lowerbound HI-STORM UMAX Weight) Cp (Carbon steel heat capacity)	 46000 kg 419 J/kg-^{o}C
Clad initial temperature (conservatively higher)	390^{o}C
Q (Decay heat)	Table 2.1.8
ΔT (clad temperature margin)	180^{o}C

Table 4.6.9

MAXIMUM COMPONENT TEMPERATURES DURING FLOOD ACCIDENT

Component	**CASE 1** Note 1 **°C (°F)**	**CASE 2** Note 1 **°C (°F)**
Fuel Cladding	368 (694)	389 (732)
MPC Basket	354 (669)	376 (709)
Basket Periphery	293 (559)	314 (597)
Aluminum Basket Shims	273 (523)	293 (559)
MPC Shell	239 (462)	260 (500)
MPC Lid*	244 (471)	264 (507)
Divider Shell	176 (349)	208 (406)
Containment Shell	56 (133)	72 (162)
Closure Lid Concrete†	108 (226)	140 (284)
Insulation	176 (349)	208 (406)

Note 1: Case 1 and Case 2 are defined in Section 4.6.2.5 as flood events of different heights to block air flow up to pedestal height and MPC baseplate respectively.

* Maximum section average temperature reported.
† Maximum section average temperature reported.

Table 4.6.10

ACCIDENT CONDITION MAXIMUM MPC PRESSURES

Condition	**Gauge Pressure (psig)**
Extreme Ambient Temperature	99.0
32 hours 100% blockage of air inlet	124.8
Burial under Debris @ Maximum Allowable Burial Time	134.6
Flood Case 1 *	95.0
Flood Case 2 *	99.0

* Flood scenario defined in Section 4.6.2.5.

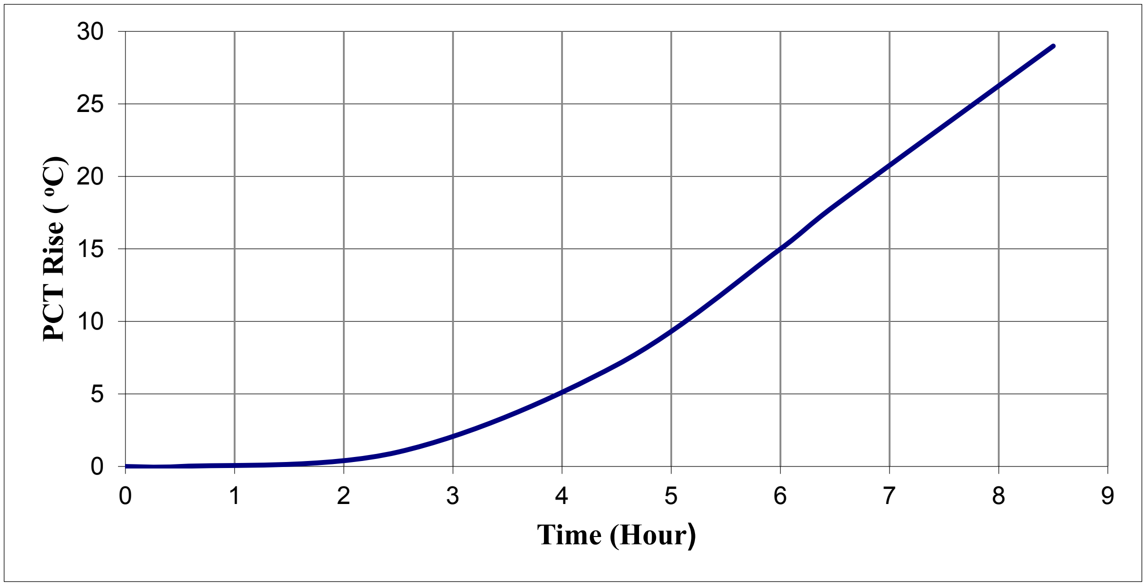

Figure 4.6.1: Rise of PCT during All Inlet Duck Blockage Accident Condition

4.7 REGULATORY COMPLIANCE

The thermal compliance pursuant to the provisions of NUREG [4.0.1] and ISG-11 [4.0.2] of the MPCs requested for certification (MPC-37 and MPC-89) in the HI-STORM UMAX system has been considered in this chapter. NUREG-1536 [4.0.1] and ISG-11 [4.0.2] define several thermal acceptance criteria that must be applied to evaluations of normal conditions of storage. These items are addressed in Sections 2.5 and 4.1. Each of the pertinent criteria and the conclusion of the evaluations are summarized here.

As required by ISG-11 [4.0.2], the fuel cladding temperature at the beginning of dry storage is maintained below the anticipated damage-threshold temperatures for normal conditions for the licensed life of the HI-STORM UMAX System. Maximum fuel cladding temperatures for long-term storage conditions are reported in Section 4.4.

As required by NUREG-1536 [4.0.1], the maximum internal pressure of the canister remains within its design pressure for normal conditions, assuming rupture of 1 percent of the fuel rods. Assumptions for pressure calculations include release of 100 percent of the fill gas and 30 percent of the significant radioactive gases in the fuel rods. Maximum internal pressures are reported in Section 4.4 and shown to remain below the normal design pressures specified in Table 2.3.5.

As required by NUREG-1536 [4.0.1], all VVM components and fuel materials are maintained within their minimum and maximum temperature for normal and off-normal conditions in order to enable components to perform their intended safety functions. Maximum and minimum temperatures for normal, off-normal and accident long-term storage conditions are reported in Sections 4.4 and 4.6 which are shown to be well below their respective Design temperature limits summarized in Table 2.3.7.

As required by NUREG-1536 [4.0.1], the system ensures a very low probability of cladding breach during long-term storage. For long-term normal conditions, the maximum CSF cladding temperature is shown to be below the ISG-11 [4.0.2] limit of 400°C (752°F).

As required by NUREG-1536 [4.0.1], the system is passively cooled. All heat rejection mechanisms described in this chapter, including conduction, natural convection, and thermal radiation, are completely passive.

As required by NUREG-1536 [4.0.1], the thermal performance of the system is within the allowable design criteria specified in SAR Chapters 2 and 3 for normal conditions. All thermal results reported in Section 4.4 are within the design criteria under all normal conditions of storage.

The thermal compliance of short term operations (such as MPC drying, lid welding and transport on the haul path) is presented in Section 4.5 wherein complete compliance with the provisions of

ISG-11 [4.0.2] is demonstrated. In particular, the ISG-11 requirement to ensure that maximum cladding temperatures under all fuel loading and short-term operations be below 400°C (752°F) for high burnup fuel and below 570°C (1058°F) for moderate burnup fuel is demonstrated.

As required by NUREG-1536 [4.0.1], the maximum internal pressure of the MPC is evaluated and shown to remain within its off-normal and accident design pressure, assuming rupture of 10 percent and 100 percent of the fuel rods, respectively. Assumptions for pressure calculations include release of 100 percent of the fill gas and 30 percent of the significant radioactive gases in the fuel rods.

It is therefore concluded that all applicable regulatory requirements and guidelines germane to the integrity of the stored fuel and the "UMAX" storage system have been addressed and satisfied in this chapter.

4.8 REFERENCES

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CHAPTER 5: SHIELDING EVALUATION OF THE HI-STORM UMAX SYSTEM

5.0 INTRODUCTION

This chapter contains a shielding safety analysis of the loaded HI-STORM UMAX VVMs pursuant to the guidelines in NUREG-1536 [5.0.1]. The objective of the analyses summarized in this chapter is to demonstrate that the loaded underground VVM will provide sufficient dose blockage to enable an on-site ISFSI to be operated at a fraction of the controlled area boundary dose limits in 10CFR72**.** The analyses presented in this chapter focus on two representative multipurpose canisters, viz. MPC-37 and MPC-32, out of the population of MPCs listed in the Table 1.2.1. A technical justification of the basis for selecting these two representative canisters is provided later in this section. This chapter, however, only supports the certification of the MPC-37 and MPC-89. The analyses reported for the smaller diameter canister are for reference purposes only. The sections that follow demonstrate that the design of the HI-STORM UMAX dry cask storage system fulfills the following acceptance criteria outlined in the Standard Review Plan, NUREG-1536:

Acceptance Criteria

1. The minimum distance from each spent fuel handling and storage facility to the controlled area boundary must be at least 100 meters. The “controlled area” is defined in 10CFR72.3 as the area immediately surrounding an ISFSI or monitored retrievable storage (MRS) facility, for which the licensee exercises authority regarding its use and within which ISFSI operations are performed.

2. The system designer must show that, during both *normal operations and anticipated occurrences*, the radiation shielding features of the proposed dry cask storage system are sufficient to meet the radiation dose requirements in Section 72.104(a). Specifically, the vendor must demonstrate this capability for a typical array of casks in the most bounding site configuration. For example, the most bounding configuration might be located at the minimum distance (100 meters) to the controlled area boundary, without any shielding from other structures or topography.

3. Dose rates from the cask must be consistent with a well-established “as low as reasonably achievable” (ALARA) program for activities in and around the storage site.

4. After a design-basis accident, an individual at the boundary or outside the controlled area shall not receive a dose greater than the limits specified in 10CFR72.106.

5. The proposed shielding features must ensure that the dry cask storage system meets the regulatory requirements for occupational and radiation dose limits for individual members of the public, as prescribed in 10CFR Part 20, Subparts C and D.

Consistent with the guidelines provided in the Standard Review Plan, NUREG-1536, this chapter contains the following information:

- A description of the shielding features of the HI-STORM UMAX System.
- A description of the source terms.
- A general description of the shielding analysis methodology.
- A description of the analysis assumptions and results for the HI-STORM UMAX system.
- Analyses to show that the 10CFR72.106 controlled area boundary radiation dose limits are met during accident conditions of storage for non-effluent radiation at a minimum distance of 100 meters.

Since only representative dose rate values for normal conditions can be presented herein, compliance with the radiation and exposure objectives of 10CFR72.104, of necessity, must be performed as part of the site specific evaluations.

In this chapter, analyses are presented for two representative MPCs, namely MPC-32 and MPC-37, previously used to qualify peer-certified HI-STORM systems (MPC-32 in HI-STORM 100 in Docket Number 72-1014 [5.0.2], and MPC-37 in HI-STORM FW in Docket Number 72-1032 [5.0.3]) showing that the radiation dose rates follow As-Low-As-Reasonably-Achievable (ALARA) principle. MPC-32 and MPC-37 are selected as they represent two different classes of MPCs from the outer dimensions perspective. The smaller diameter of the MPC-32 compared to that of the MPC-37 results in a larger gap between the MPC enclosure shell and CEC. Additionally, MPC-32 is slightly longer than MPC-37. Both of these attributes may cause slightly higher dose rates for the HI-STORM UMAX system loaded with MPC-32 compared to MPC-37. As stated at the beginning of this section, the MPC-32 dose evaluations are for reference purpose only. Besides the MPC-37, this chapter also supports the certification of the MPC-89 in the HI-STORM UMAX system. However, the MPC-89 specific shielding analyses are not performed as the results presented in the HI-STORM FW FSAR show that results for the MPC-89 are similar to those for the MPC-37. Additionally, it is noted that site specific analyses need to use the site specific MPC (MPC-37 or MPC-89) for controlled area boundary dose calculations to show the site's compliance with 10 CFR 72.104 and 10 CFR 72.106.

The design basis zircaloy clad fuel assemblies used for calculating the dose rates presented in this chapter is the Westinghouse (W) 17x17 PWR fuel (see Table 5.0.1). The burnup and cooling times used for the shielding evaluation are consistent[‡‡] with those in [5.0.3]. Use of identical MPC and similar source term data enables a one-to-one comparison between the

[‡‡] Only difference is the cooling time: FW uses 4.5 years which is more representative for the side of the MPC, whereas UMAX uses 5 years which is more representative for the top of the MPC.

shielding capacity of "HI-STORM UMAX" and its aboveground counterpart, the HI-STORM FW MPC storage system.

The principal sources of radiation in the HI-STORM UMAX System are:

- Gamma radiation originating from the following sources:
  1. Decay of radioactive fission products
  2. Secondary photons from neutron capture in fissile and non-fissile nuclides
  3. Hardware activation products generated during core operations

- Neutron radiation originating from the following sources:
  1. Spontaneous fission
  2. α,n reactions in fuel materials
  3. Secondary neutrons produced by fission from subcritical multiplication
  4. γ,n reactions (this source is negligible)

10CFR72 contains two sections that set down main dose requirements: §104 for normal and off-normal conditions, and §106 for anticipated occurrences and accident conditions. The relationship of these requirements to the safety analyses documented herein, are as follows:

- 10CFR72.104 specifies the dose limits from an ISFSI (and other operations) at a site boundary under normal and off-normal conditions. Compliance with §104 can therefore only be demonstrated on a site-specific basis, since it depends not only on the design of the cask system and the loaded fuel, but also on the ISFSI layout, the distance to the site boundary, and possibly other factors such as the terrain around the ISFSI. The purpose of this chapter is therefore to present the analysis methodology and illustrate its application to obtain the dose rates at locations of interest using a reference problem. The analysis carried out on the "HI-STORM UMAX" geometry is also intended to aid the user in applying the ALARA considerations and planning of the ISFSI. To accomplish the above objectives, it is appropriate to present reasonably conservative dose rate values, based on a reasonable conservative choice of burn-ups and cooling times of the assemblies (Table 5.0.1).
- For the accident dose limit in 10CFR72.106 it should be noted that the governing accident condition for the HI-STORM FW system, namely loss of shielding water in the Transfer cask, is analyzed in Reference [5.0.3]. Reference [5.0.3] contains the shielding analysis for accidental loss of water in HI-TRAC VW. Likewise, the HI-STORM 100 docket (Docket Number 72-1014) contains the equivalent accident analysis for HI-TRAC 125 and 100 models used to transfer MPC-32, MPC-68, and other models licensed in [5.0.2], which can be used as references for ISFSI planning. Therefore, the classical accident analysis considered in deploying the HI-STORM system are pre-empted by the evaluations in Reference [5.0.3], making it unnecessary and redundant to perform them in this FSAR. In addition, a Design Basis Missile (defined by Tables 2.3.3 and 2.3.4) strike to the exposed radiation protection space boundary (defined in Chapter 1), created by the excavation adjacent to it, has been analyzed in Chapter 3 and found to cause about 5.5 feet penetration

through the subgrade. It is to be noted that the optional enclosure wall was not considered in the missile penetration calculations, and while the radiation protection space provides at least 10.75 feet of subgrade in the radial direction from the outer metal surface of the VVM to the radiation protection space boundary, only 6.5 feet of subgarde is conservatively utilized for this accident condition dose calculation. Hence, the impact of a tornado missile penetrating a 6.5 feet subgrade creating a horizontal hole extending 5.5 feet from the external surface of the subgrade is considered as an accident condition for the shielding evaluations.

The shielding analyses were performed with MCNP5 [5.0.4] developed by Los Alamos National Laboratory (LANL). The source terms for the design basis fuels were calculated with the SAS2H and ORIGEN-S sequences from the SCALE 5 system [5.0.5][5.0.6]. A detailed description of the MCNP models and the source term calculations are presented in Sections 5.3 and 5.2, respectively.

To facilitate convenient access to the referenced material, the latest edition of the HI-STORM FW FSAR has been placed in this docket and a list of "FW" FSAR sections germane to this chapter is provided in a tabular form. The HI-STORM FW FSAR will be maintained in a configuration controlled status in this docket as a mandatory supplement to this FSAR. The table below provides a listing of the material adopted in this chapter by reference to the HI-STORM FW FSAR.

The safety analyses summarized in this chapter demonstrate that under accident conditions, acceptable margins to allowable limits exist under all design basis loading conditions. For normal and off-normal conditions, the analyses in this chapter simply provide a generic evaluation that demonstrates that the dose requirements as specified in 10CFR72.104 can be met under site specific conditions. Minor changes to the design parameters that inevitably occur during the product’s life cycle which are treated within the purview of 10CFR72.48 and are ascertained to have an insignificant effect on the computed dose rates in this chapter may not prompt a formal reanalysis and revision of the results and associated data in the tables of this chapter unless the cumulative effect of all such unquantified changes cannot be deemed any more to be insignificant. For accident conditions, the dose limit as specified in 10CFR72.106 is 5 rem. The only accident which impacts dose rates is the loss of water in the water jacket for the HI-TRAC VW. For the purposes of determining if the changes to the HI-TRAC VW are insignificant, an insignificant loss of margin with reference to the 5 rem acceptance criteria is defined as the estimated reduction that is no more than one order of magnitude less than the available margin reported in the FSAR. For normal and off-normal conditions, site specific dose evaluations are required to demonstrate compliance with 10CFR72.104. Incorporating any minor changes into those site specific evaluations is only warranted if it would be expected, on a site specific basis, that those changes could result in a situation where the limits are no longer met and where therefore other compensatory measures are required, such as a change in the loading plan or the closure lid concrete density. Incorporating changes into the analyses in this chapter for normal and off-normal conditions will only be performed under extenuating circumstances, e.g. major changes to the shielding design, in order to provide an updated template for the site specific dose analyses.

To ensure rigorous configuration control, the information in the Licensing drawings in Section 1.5 should be treated as the authoritative source for numerical analysis at all times. Reliance on the input data and associated results in this chapter for additional mathematical computations may not be appropriate as they serve the sole purpose of establishing safety compliance in accordance with the acceptance criteria set down in Chapter 2 and in this chapter.

SECTIONS OF HI-STORM FW FSAR APPLICABLE TO THIS SAFETY EVALUATION [§§]		
Location in the UMAX FSAR	**Subject of the Reference**	**Location in the HI-STORM FW FSAR, Revision 3**
Sections 5.0 and Sub-Section 5.1.1	Shielding safety analyses governing the HI-TRAC	Section 5.1 and 5.4
Section 5.2	The reference fuel and the Non-fuel hardware (BPRAs etc)	Section 5.2
Section 5.4	The axial distribution of the fuel source term	Table 2.1.5 Figures 2.1.3 and 2.1.4

[§§] For convenience of reference, the specific revision of the HI-STORM FW FSAR that is referenced in the safety analysis herein is placed in this docket. Updated versions of the HI-STORM FW FSAR shall be placed in this docket as necessary so as to ensure that the safety analyses on the "UMAX" docket (72-1040) remain aligned with the material referenced in the HI-STORM FW FSAR.

Table 5.0.1

DESIGN BASIS FUEL BURNUP, COOLING TIME, AND ENRICHMENT FOR DOSE EVALUATION

MPC TYPE	BURN- UP GWD/MTU	COOLING TIME YEARS	ENRICHMENT Wt % U-235
MPC-32 and MPC-37	45	5	3.6

5.1 SHIELDING FEATURES, DESIGN OBJECTIVE AND RESULTS

5.1.1 Shielding Features

The essentials of the HI-STORM UMAX System are described in Section 1.2. The design details are laid out in the licensing drawings in Section 1.5. As can be seen from the Licensing drawings, the underground HI-STORM UMAX System differs from all aboveground HI-STORM overpacks certified in Dockets 72-1014 and 1032 in that the used fuel is stored well below the ISFSI's top of grade (TOG). HI-STORM UMAX, however, is completely fungible with the HI-STORM FW overpack model in that it can store the MPC-37 and MPC-89 certified to be stored in the FW system. Furthermore, the HI-TRAC VW transfer cask used to perform the *short-term operations* to prepare and install the MPCs in the storage system is identical to that used in the HI-STORM FW system [5.0.3]. In fact, the MPC's content conditions and the loading operations up to the time the loaded transfer cask arrives at the ISFSI are identical to the HI-STORM FW system. Therefore, all shielding safety analyses governing the HI-TRAC VW transfer cask are already provided in Reference [5.0.3] and no further calculations involving the HI-TRAC VW transfer cask are necessary or presented in this chapter.

As shown in the Licensing drawings, the HI-STORM UMAX ISFSI consists of a set of vertically disposed thick-walled steel containers founded on a thick reinforced concrete pad (denoted as the Support Foundation pad, located over 20 feet below TOG) and embedded in a subgrade made of a Self-hardening Engineered Subgrade. The top region of the steel container is reinforced by a thick plate-type flange that rests on a reinforced concrete pad denoted as the ISFSI pad. The top opening in the container is the only location of access into the cavity and potential path of emission of radiation to the environment. To provide maximum blockage to the radiation issuing from the fuel, "HI-STORM UMAX" utilizes both mass and distance as barriers. The Closure Lid, made as an over-40 inch thick steel weldment filled with concrete, provides a massive body in the path of the radiation emanating from the fuel. The shielding action of the lid is further aided by ensuring that the top of the MPC (itself equipped with a > 9inch thick lid) is at least about 2 feet below the bottom of the VVM Closure lid. With such a massive blockage arrayed against the stored fuel, the dose levels at the top of the ISFSI pad are expected to be very low, which the analyses in this chapter will be seen to corroborate.

However, aside from the magnitude of shielding, the shielding mechanism in the HI-STORM UMAX VVM is similar to the aboveground overpack designs certified in [5.0.2] and [5.0.3] with gamma shielding provided by the concrete and the steel of the module, and neutron shielding provided by the ISFSI concrete. However, because the VVM is located below grade, a significant reduction in the dose at the directly accessible surface of the VVM compared to the aboveground storage modules is realized. Dose rates from a HI-STORM UMAX VVM at the site boundary are therefore significantly lower than, and bounded by, the corresponding dose rates from the peer aboveground HI-STORM systems.

It should be noted that the material and thickness of some parts are optional in the HI-STORM UMAX drawings provided in Chapter 1. Site specific chosen options will be considered in performing site specific shielding calculations to show the compliance with the regulatory limits.

5.1.2 Design Objective

In accordance with ALARA practices, design objective dose rates are established for the HI-STORM UMAX system and presented in Table 2.9.2.

Chapter 12 discusses the potential off-normal conditions and their effect on the HI-STORM UMAX system. None of the off-normal conditions have any impact on the shielding analysis. Therefore, off-normal and normal conditions are identical for the purpose of the shielding evaluation.

The 10CFR72.104 criteria for radioactive materials in effluents and direct radiation during normal operations are:

1. During normal operations and anticipated occurrences, the annual dose equivalent to any real individual who is located beyond the controlled area, must not exceed 25 mrem to the whole body, 75 mrem to the thyroid, and 25 mrem to any other critical organ.

2. Operational restrictions must be established to meet As-Low-as-Reasonably-Achievable (ALARA) objectives for radioactive materials in effluents and direct radiation.

10CFR20 Subparts C and D specify additional requirements for occupational dose limits and radiation dose limits for individual members of the public. Chapter 11 specifically addresses these regulations.

5.1.3 Results

Figure 5.1.1 identifies the locations of the dose points referenced in Tables 5.1.1 and 5.1.2 for HI-STORM UMAX loaded with MPC-32 and MPC-37, respectively. Dose Points #1 represents the side of the closure lid shell on top of the inlet plenum. Dose Point #1 is also the location of the highest dose rate on the lid in the final storage configuration. However, there is a substantial dose rate reduction at 1 meter from Dose Point #1. The maximum dose rate is reported for the side surface of the lid shell, while the dose rate value reported at 1 meter is taken at the middle of the lid shell. Dose Point #2 is the location of the surface of the outlet duct. Dose Point #3 is positioned on the closure lid cover plate. Dose Points # 4 and #5 (#5 not seen in Figure 5.1.1) are the locations of the outlet and inlet vents (top surface), respectively. Dose Point #6 is located over a tube that would be required for the ICCPS test station if an ICCPS is used. Dose Point #7 is located over an empty VVM located adjacent to a loaded VVM. Dose Point #7 is used to calculate the potential radiation streaming through an empty cavity surrounded by 4 loaded VVMs.

The tube for the ICCPS test station is modeled as a cylindrical hole that extends from the VIP down to the base plate of the MPC. The tube is modeled with a diameter of 4 inches, located about 5.5 feet from the center of the VVM. If the actual tube has characteristics that could result in higher dose rates, i.e. is larger or closer to the VVM than modeled here, the actual tube

characteristics should be considered in the site specific dose calculations. Depending on the results of those calculations, additional measures, such as added shielding at the top of the tube, may be required.

A comparison between the dose rates in Table 5.1.2 and the corresponding dose rates presented in Section 5.1 of the HI-STORM FW FSAR [5.0.3] show that the maximum dose rate for the HI-STORM UMAX module with a loaded MPC is well below the maximum dose rate for the HI-STORM FW with the identically loaded MPC. The generally lower dose rate in HI-STORM UMAX compared to HI-STORM FW would suggest that the dose rate contribution at the site boundary will be accordingly exiguous. Nevertheless, calculations were performed for both MPC-32 and MPC-37 under normal condition to determine the annual dose rate from the HI-STORM UMAX system at a distance of 100 meters. These results, which are presented in Table 5.1.3, indicate that the HI-STORM UMAX meets the requirements of 10CFR72.104 at 100 meters with large margins. Comparing these results to the results in Section 5.1 of the HI-STORM FW FSAR demonstrates that the off-site dose from the HI-STORM UMAX is a small fraction of the off-site dose from an aboveground overpack.

The bounding accident condition is identified in Chapter 3 to be the impact of a tornado missile with a diameter of 8 inches. This missile would penetrate the soil about 5.5 ft. This is the bounding condition since smaller missiles have less energy, and larger missiles (automotive) have a much larger impact area thus resulting in a much smaller indentation of the soil. Under the bounding condition, the maximum dose over a period of 30 days at a distance of 100 m from the VVM is presented in Table 5.1.4. As discussed in Section 5.0, only 6.5 feet of subgrade in the radial direction from the outer metal surface of the VVM is applied for this accident condition dose evaluation. Table 5.1.4 demonstrates that the accident condition dose is in compliance with the 10CFR72.106 and much lower than the dose reported for HI-TRAC accident condition in Reference [5.0.3].

Table 5.1.1

DOSE RATES ADJACENT TO AND 1 METER FROM THE HI-STORM UMAX MODULE FOR NORMAL CONDITIONS
MPC-32 DESIGN BASIS ZIRCALOY CLAD FUEL

Dose Point† Location	**Fuel Gammas†† (mrem/hr)**	**^{60}Co Gammas (mrem/hr)**	**Neutrons (mrem/hr)**	**Totals (mrem/hr)**	**Totals with BPRAs (mrem/hr)**
Surface of the overpack					
1	22.37	12.71	21.80	**56.88**	**66.04**
2	0.60	0.07	2.36	**3.03**	**3.10**
3	0.17	0.01	0.53	**0.72**	**0.73**
4	1.35	0.22	0.81	**2.37**	**2.55**
5	3.69	2.02	2.95	**8.65**	**10.11**
6	10.66	4.93	1.31	**16.90**	**20.28**
7‡	1.04	0.48	0.76	**2.24**	**2.56**
One meter from the overpack					
1	5.67	1.43	1.19	**8.29**	**9.56**
2	3.34	1.62	1.95	**6.91**	**8.14**
3	0.26	0.05	0.22	**0.53**	**0.57**
4	0.42	0.07	0.44	**0.93**	**0.99**
5	0.71	0.34	0.43	**1.48**	**1.72**

† Refer to Figure 5.1.1.
†† Gammas generated by neutron capture are included with fuel gammas.
‡ Calculated for an empty VVM surrounded by four loaded VVMs.

Table 5.1.2

DOSE RATES ADJACENT TO AND 1 METER FROM THE HI-STORM UMAX MODULE FOR NORMAL CONDITIONS
MPC-37 DESIGN BASIS ZIRCALOY CLAD FUEL

Dose Point† Location	Fuel Gammas†† (mrem/hr)	^{60}Co Gammas (mrem/hr)	Neutrons (mrem/hr)	Totals (mrem/hr)	Totals with BPRAs (mrem/hr)
Surface of the overpack					
1	15.67	4.99	17.28	**37.93**	**41.47**
2	0.52	0.04	1.76	**2.32**	**2.35**
3	0.15	0.01	0.42	**0.57**	**0.57**
4	1.15	0.15	0.64	**1.94**	**2.08**
5	2.20	0.76	2.27	**5.24**	**5.77**
6	6.00	4.11	1.15	**11.27**	**13.87**
7‡	1.88	0.2	0.56	**2.64**	**2.8**
One meter from the overpack					
1	3.81	0.80	0.96	**5.58**	**6.14**
2	2.30	0.63	1.53	**4.45**	**4.92**
3	0.26	0.02	0.18	**0.47**	**0.53**
4	0.36	0.05	0.35	**0.76**	**0.80**
5	0.64	0.15	0.36	**1.14**	**1.27**

† Refer to Figure 5.1.1.
†† Gammas generated by neutron capture are included with fuel gammas.
‡ Calculated for an empty VVM surrounded by four loaded VVMs.

Table 5.1.3 ANNUAL DOSE AT 100 METERS FROM A SINGLE HI-STORM UMAX OVERPACK WITH MPC-32 AND MPC-37 WITH DESIGN BASIS ZIRCALOY CLAD FUEL†		
Dose Component	**MPC-32 Dose Rates (mrem/yr)**	**MPC-37 Dose Rates (mrem/yr)**
Fuel gammas††	3.85	2.80
^{60}Co Gammas	1.23	0.61
Neutrons	2.54	2.10
Totals	7.62	5.43
Totals with BPRAs	**8.58**	**5.96**

† 8760 hour annual occupancy is assumed.

†† Gammas generated by neutron capture are included with fuel gammas.

Table 5.1.4 DOSE AT 100 METERS FROM A SINGLE HI-STORM UMAX OVERPACK WITH MPC-32 AND MPC-37 LOADED WITH DESIGN BASIS FUEL FOR ACCIDENT CONDITION	
MPC	**DOSE§ (Rem)**
MPC-32	**0.12**
MPC-37	**0.13**

§ Accident duration is assumed to be 30 days.

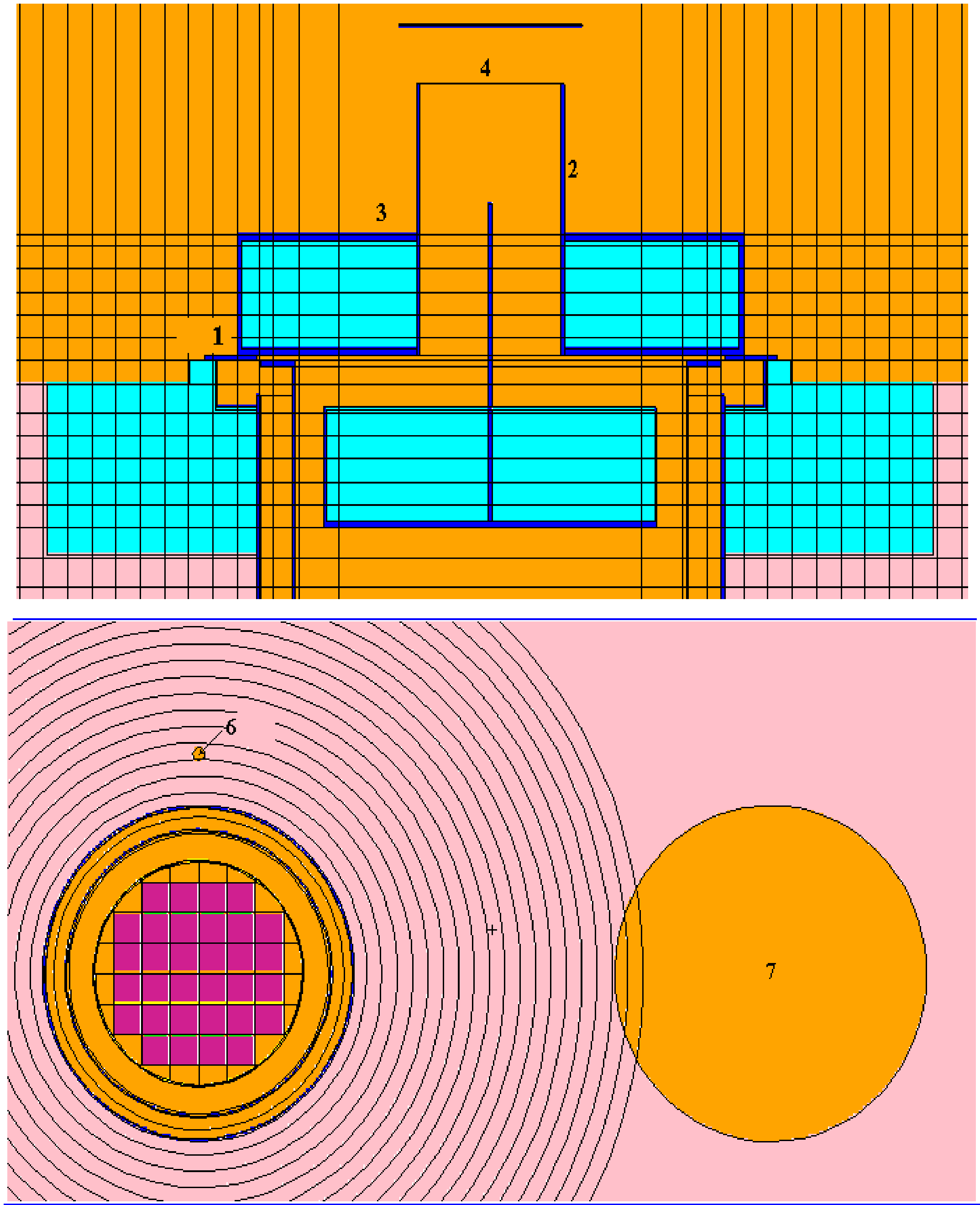

FIGURE 5.1.1: HI-STORM UMAX MODULE CROSS SECTIONAL VIEWS WITH DOSE POINT LOCATIONS
(SKY COLOR REPRESENTS CONCRETE, PINK REPRESENTS SUBGRADE AND ORANGE REPRESENTS AIR)

5.2 SOURCE SPECIFICATION

The design basis fuel used for the reference shielding analyses is identical, and the burnup, cooling time and enrichment combination is consistent** with what was used in the HI-STORM FW FSAR [5.0.3]. However, to enhance readability and to assist the reviewer, excerpt from the source specification section, Section 5.2 of Chapter 5, of the HI-STORM FW FSAR is presented here.

††The neutron and gamma source terms, decay heat values, and quantities of radionuclides available for release were calculated with the SAS2H and ORIGEN-S modules of the SCALE 5 system [5.0.5][5.0.6]. SAS2H has been extensively compared to experimental isotopic validations and decay heat measurements. References [5.2.1] through [5.2.8] present isotopic and decay heat comparisons for PWR and BWR fuels. All of these studies indicate good agreement between SAS2H and ORIGEN-S and measured data.

A description of the design basis fuel for the source term calculations is provided in Table 5.2.1. Subsection 5.2.4 discusses the rationale in the determination of the design basis fuel assemblies.

In performing the SAS2H and ORIGEN-S calculations, a single full power cycle was used to achieve the desired burnup. This assumption, in conjunction with the above-average specific powers listed in Table 5.2.1 resulted in a conservative source term calculation.

5.2.1 Gamma Source

Tables 5.2.2 provides the gamma source in MeV/s and photons/s as calculated with SAS2H and ORIGEN-S for the design basis zircaloy clad fuel at the burnups and cooling times used for the shielding analyses in this chapter.

Previous analyses were performed for the HI-STORM 100 system to determine the dose contribution from gammas as a function of energy [5.0.2]. The results of these analyses have revealed that, due to the magnitude of the gamma source at lower energies, photons with energies as low as 0.45 MeV must be included in the shielding analysis, but photons with energies below 0.45 MeV are too weak to penetrate the HI-STORM overpack or HI-TRAC. The effect of gammas with energies above 3.0 MeV, on the other hand, was found to be insignificant. This is due to the fact that the source of

** Only difference is the cooling time: FW uses 4.5 years which is more representative for the side of the MPC, whereas UMAX uses 5 years which is more representative for the top of the MPC.
†† The text matter in the "Arial" font is excerpted from the HI-STORM FW FSAR with minor editorial changes, as appropriate.

gammas in this range (i.e., above 3.0 MeV) is extremely low. Therefore, all photons with energies in the range of 0.45 to 3.0 MeV are included in the shielding calculations.

The primary source of activity in the non-fuel regions of an assembly arises from the activation of ^{59}Co to ^{60}Co. The primary source of ^{59}Co in a fuel assembly is impurities in the steel structural material above and below the fuel. The zircaloy in these regions is neglected since it does not have a significant ^{59}Co impurity level. Reference [5.2.9] indicates that the impurity level in steel is 800 ppm or 0.8 gm/kg. Therefore, inconel and stainless steel in the non-fuel regions are both assumed to have the same 0.8 gm/kg impurity level.

Some of the PWR fuel assembly designs (B&W and WE 15x15) utilized inconel in-core grid spacers while other PWR fuel designs use zircaloy in-core grid spacers. In the mid 1980s, the fuel assembly designs using inconel in-core grid spacers were altered to use zircaloy in-core grid spacers. To take that into account, the gamma source for the PWR zircaloy clad fuel assembly includes the activation of the in-core grid spacers.

The masses in Table 5.2.1 were used to calculate a ^{59}Co impurity level in the fuel assembly material. The grams of impurity were then used in ORIGEN-S to calculate a ^{60}Co activity level for the desired burnup and decay time. The methodology used to determine the activation level was developed from Reference [5.2.10] and is described here.

1. The activity of the ^{60}Co is calculated using ORIGEN-S. The flux used in the calculation was the in-core fuel region flux at full power.

2. The activity calculated in Step 1 for the region of interest was modified by the appropriate scaling factors listed in Table 5.2.3. These scaling factors were taken from Reference [5.2.10].

Table 5.2.4 presents the ^{60}Co activity utilized in the shielding calculations for the non-fuel regions of the assemblies in the MPC-37 and the MPC-32.

In addition to the two sources already mentioned, a third source arises from (n,γ) reactions in the material of the MPC and the overpack. This source of photons is properly accounted for in MCNP when a neutron calculation is performed in a coupled neutron-gamma mode.

5.2.2 Neutron Source

It is well known that the neutron source strength increases as enrichment decreases, for a constant burnup and decay time. This is due to the increase in Pu content in the fuel, which increases the inventory of other transuranium nuclides such as Cm. The gamma source also varies with enrichment, although only slightly. Because of this effect and in

order to obtain conservative source terms, low initial fuel enrichments of 3.6 wt% were chosen for the PWR design basis fuel assemblies under.

The neutron sources calculated for the design basis fuel assembly are listed in Tables 5.2.5 in neutrons/s for the selected burnup and cooling times used in the shielding evaluations. The neutron spectrum is generated in ORIGEN-S.

5.2.3 Non-Fuel Hardware

Burnable poison rod assemblies (BPRAs), thimble plug devices (TPDs), control rod assemblies (CRAs), axial power shaping rods (APSRs), and neutron source assemblies (NSAs)) are generally termed as non-fuel hardware and can be stored as an integral part of a PWR fuel assembly. Non-fuel hardware storage restrictions as applicable for a particular MPC in the certification of its host docket are also applicable in the HI-STORM UMAX system. Similar to HI-STORM FW FSAR [5.0.3], representative shielding analyses are performed in this chapter by only utilizing BPRAs in every PWR fuel location in the basket.

5.2.3.1 BPRAs

Burnable poison rod assemblies (BPRA) (including wet annular burnable absorbers) are an integral, yet removable, part of a large portion of PWR fuel. BPRAs are made of stainless steel in the region above the active fuel zone and may contain a small amount of inconel in this region. Within the active fuel zone the BPRAs may contain 2-24 rodlets which are burnable absorbers clad in either zircaloy or stainless steel. The stainless steel clad BPRAs create a significant radiation source (Co-60) while the zircaloy clad BPRAs create a negligible radiation source. Therefore, the stainless steel clad BPRAs are bounding.

The masses of this devise are listed in Table 5.2.6, while Table 5.2.7 presents the curies of Co-60 that were calculated for BPRAs in each region of the fuel assembly (e.g. incore, plenum, top). For specific site boundary evaluations, these levels/values can be used if they are bounding. Alternatively, more realistic values can be used.

5.2.4 Choice of Design Basis Assembly

The Westinghouse 17x17 assembly was selected as design basis assembly since it is been widely used throughout the industry. Site specific shielding evaluations should verify that those assemblies and assembly parameters are appropriate for the site-specific analyses.

Table 5.2.1
DESIGN BASIS PWR FUEL DATA

Item	Data
Assembly type/class	WE 17×17
Active fuel length (in.)	144
No. of fuel rods	264
Rod pitch (in.)	0.496
Cladding material	Zircaloy-4
Rod diameter (in.)	0.374
Cladding thickness (in.)	0.0225
Pellet diameter (in.)	0.3232
Pellet material	UO_2
Pellet density (gm/cc)	10.412 (95% of theoretical)
Enrichment (w/o ^{235}U)	3.6
Specific power (MW/MTU)	43.48
Weight of UO_2 (kg)[††]	532.150
Weight of U (kg)[††]	469.144
No. of Water Rods/ Guide Tubes	25
Water Rod/ Guide Tube O.D. (in.)	0.474
Water Rod/ Guide Tube Thickness (in.)	0.016
Lower End Fitting (kg)	5.9 (steel)
Gas Plenum Springs (kg)	1.150 (steel)
Gas Plenum Spacer (kg)	0.793 (inconel) 0.841 (steel)
Expansion Springs (kg)	N/A
Upper End Fitting (kg)	6.89 (steel) 0.96 (inconel)
Handle (kg)	N/A
Incore Grid Spacers (kg)	4.9 (inconel)

[††] Derived from parameters in this table.

Table 5.2.2

CALCULATED PWR FUEL GAMMA SOURCE PER ASSEMBLY FOR DESIGN BASIS BURNUP AND COOLING TIME

Lower Energy	**Upper Energy**	**45,000 MWD/MTU 5-Year Cooling**	
(MeV)	(MeV)	(MeV/s)	(Photons/s)
0.45	0.7	1.95E+15	3.40E+15
0.7	1.0	6.52E+14	7.67E+14
1.0	1.5	1.52E+14	1.22E+14
1.5	2.0	1.19E+13	6.79E+12
2.0	2.5	6.64E+12	2.95E+12
2.5	3.0	2.88E+11	1.05E+11
Total		2.78E+15	4.30E+15

Table 5.2.3

SCALING FACTORS USED IN CALCULATING THE ^{60}Co SOURCE

Region	**PWR**
Handle	N/A
Upper End Fitting	0.1
Gas Plenum Spacer	0.1
Expansion Springs	N/A
Gas Plenum Springs	0.2
Incore Grid Spacer	1.0
Lower End Fitting	0.2

Table 5.2.4

CALCULATED ^{60}Co SOURCE PER ASSEMBLY
FOR DESIGN BASIS
FUEL AT DESIGN BASIS BURNUP AND COOLING TIME

Location	**45,000 MWD/MTU and 5-Year Cooling (curies)**
Lower End Fitting	80.53
Gas Plenum Springs	15.70
Gas Plenum Spacer	11.15
Incore Grid Spacers	334.42
Upper End Fitting	53.57

Table 5.2.5

CALCULATED PWR NEUTRON SOURCE PER ASSEMBLY FOR 45,000 MWD/MTU BURNUP AND 5 YEAR COOLING

Lower Energy (MeV)	**Upper Energy (MeV)**	**45,000 MWD/MTU 5-Year Cooling (Neutrons/s)**
1.0e-01	4.0e-01	2.99E+07
4.0e-01	9.0e-01	6.52E+07
9.0e-01	1.4	6.51E+07
1.4	1.85	5.20E+07
1.85	3.0	9.69E+07
3.0	6.43	8.80E+07
6.43	20.0	8.40E+06
Totals		4.06E+08

Table 5.2.6 DESCRIPTION OF DESIGN BASIS BURNABLE POISON ROD ASSEMBLY	
Region	**BPRA**
Upper End Fitting (kg of steel)	2.62
Upper End Fitting (kg of inconel)	0.42
Gas Plenum Spacer (kg of steel)	0.77488
Gas Plenum Springs (kg of steel)	0.67512
In-core (kg of steel)	13.2

Table 5.2.7 DESIGN BASIS COBALT-60 ACTIVITIES FOR BURNABLE POISON ROD ASSEMBLIES	
Region	**BPRA**
Upper End Fitting (curies Co-60)	32.7
Gas Plenum Spacer (curies Co-60)	5.0
Gas Plenum Springs (curies Co-60)	8.9
In-core (curies Co-60)	848.4

5.3 MODEL SPECIFICATIONS

The shielding analyses of the HI-STORM UMAX module are performed with MCNP5 [5.0.4], which is a QA-validated code under Holtec International's quality program. A sample input file for MCNP is provided in Appendix 5.A.

Section 1.5 provides the drawings that describe the HI-STORM UMAX System. The nominal dimensions in these drawings were used to create the MCNP models used in the radiation transport calculations. Modeling deviations from these drawings are discussed below. Figure 5.3.1 shows cross sectional views of the HI-STORM UMAX module as it was modeled in MCNP for accident condition. Note that the inlet and outlet vents were modeled explicitly; therefore, streaming through these components is accounted for in the dose calculations. Figure 5.3.2 depicts the inlet plenum, part of the outlet and additionally the dose locations used to obtain a dose profile across the HI-STORM UMAX lid and surrounding ISFSI pad.

Since the HI-STORM UMAX models analyzed in this chapter use principally the same MPC models from the References [5.0.2][5.0.3], all figures, conservative modeling approximations, and modeling differences for the MPCs reported in the corresponding FSARs are applicable to the evaluations in this chapter. The differences between models and drawings for the module are listed and discussed here.

1. Minor penetrations in the body of the module (e.g. lift locations) are not modeled as these are small localized effects which will not affect the off-site dose rates.

2. The MPC supports and guides were conservatively neglected. The bottom pedestal was also neglected in the MCNP model.

3. The inlet plenum support plates and inlet plenum corner gussets were not included in the model.

4. The insulation installed on the divider shell was conservatively modeled as a void.

5. The cavities representing the optional ICCPS tube and the empty VVM are modeled as empty volumes surrounded by soil, i.e. any steel liner or other material in these areas, or any covers that would be located on top of those cavities, are conservatively neglected

6. The air inlets at the bottom of the divider shell were not modeled. This has negligible effect as most of the radiation is emanating from the top and side of the MPC.

7. The radius of the Closure Lid Upper Shield and the thickness of the Closure Lid Outer Shell are modeled as 53.625" and 1", while the actual dimensions are 54.125" and 0.5", respectively. The effect of the deviation on the dose rates is minimal since streaming through the inlet and outlet vents is the dominating

contributor to the dose rates. Also, the total thickness of the shielding materials is not changed. The actual dimensions will be considered for the site specific calculations.

8. The concrete density of the SFP used in the model is 2.3 g/cm^3, which is more than the actual concrete density; however, the impact on the dose rate is insignificant. The concrete density of 2.4 g/cm^3 is used in the shielding analysis for part of the ISFSI Pad (the soil density of 1.7 g/cm^3 (106 lb/ft^3) is used for the remainder of the ISFSI Pad). This concrete density may be more than the actual density of the soil/concrete of the ISFSI Pad. The actual density will be considered for the site specific calculations.

5.3.1 Fuel Configuration

The active fuel region is modeled as a homogenous zone. Calculations were performed for the HI-STORM 100 [5.0.2] to determine the acceptability of homogenizing the fuel assembly versus explicit modeling. Based on these calculations it was concluded that it is acceptable to homogenize the fuel assembly without loss of accuracy. The width of the PWR homogenized fuel assembly is equal to 17 times the pitch. The end fittings and the plenum regions are also modeled as homogenous regions of steel. The masses of steel used in these regions are shown in Table 5.2.1. The axial description of the design basis fuel assemblies is provided in Table 5.3.1.

5.3.2 Regional Densities

Composition and densities of the various materials used in the HI-STORM UMAX system for shielding analyses are given in Table 5.3.2. All of the materials and their actual geometries are represented in the MCNP model.

5.3.3 HI-STORM UMAX Optional Features

The HI-STORM UMAX system utilizes some items with optional specifications, based on site specific projects. Table 5.3.3 provides material and thickness of these items, as used in the shielding analyses. As stated in Subsection 5.1.1, site specific chosen options will be considered in performing site specific shielding calculations to show the compliance with the regulatory limits.

Also, to further reduce the occupational and site boundary dose rates, an optional divider shell shield ring (attached to the divider shell) can be implemented. The drawings provided in Chapter 1 give more information about this shield ring specification. The results presented in this chapter are for the cases without this shield ring. However, in site specific dose evaluations, this shield ring may be credited if it is used.

Table 5.3.1

DESCRIPTION OF THE AXIAL MCNP MODEL OF THE FUEL ASSEMBLIES†

Region	Start (in.)	Finish (in.)	Length (in.)	Actual Material	Modeled Material
PWR					
Lower End Fitting	0.0	2.738	2.738	SS304	SS304
Space	2.738	3.738	1.0	zircaloy	void
Fuel	3.738	147.738	144.0	fuel & zircaloy	fuel & zircaloy
Gas Plenum Springs	147.738	151.916	4.178	SS304 & inconel	SS304
Gas Plenum Spacer	151.916	156.095	4.179	SS304 & inconel	SS304
Upper End Fitting	156.095	159.765	3.670	SS304 & inconel	SS304

† All dimensions start at the bottom of the fuel assembly. The length of the fuel shims must be added to the distances to determine the distance from the top of the MPC baseplate.

Table 5.3.2

COMPOSITION OF THE MATERIALS IN THE HI-STORM FW SYSTEM

Component	Density (g/cm³)	Elements	Mass Fraction (%)
Metamic	2.642	B-10	4.388
		B-11	20.436
		Al	68.275
		C	6.901
Metamic-HT	[PROPRIETARY INFORMATION WITHHELD IN ACCORDANCE WITH 10 CFR 2.390]		
Carbon steel	7.82	Fe	99.0
		C	1.0
SS304	7.94	Cr	19.0
		Mn	2.0
		Fe	69.5
		Ni	9.5
Concrete	2.4 (150 lb/ft3) (See Note 1)	O	53.2
		Si	33.7
		Ca	4.4
		Al	3.4
		Na	2.9
		Fe	1.4
		H	1.0
Soil	1.7	H	0.962
		O	54.361
		Al	12.859
		Si	31.818

Note 1: The concrete density may be less than the value indicated here, as discussed in Section 5.3.

Table 5.3.2 (continued)

COMPOSITION OF THE MATERIALS IN THE HI-STORM FW SYSTEM

Component	Density (g/cm^3)	Elements	Mass Fraction (%)
PWR Fuel Region Mixture	3.769 (5.0 wt% U-235)	^{235}U	3.709
		^{238}U	70.474
		O	9.972
		Zr	15.565
		Cr	0.016
		Fe	0.033
		Sn	0.230
Lower End Fitting (PWR)	1.849	SS304	100
Gas Plenum Springs (PWR)	0.23626	SS304	100
Gas Plenum Spacer (PWR)	0.33559	SS304	100
Upper End Fitting (PWR)	1.8359	SS304	100

Table 5.3.3
MATERIAL AND THICKNESS OF ITEMS WITH OPTIONAL SPECIFICATIONS, AS USED IN THE SHIELDING ANALYSES

Part Name	**Material**	**Thickness (inches)**
CEC Containment Shell	Carbon steel	0.75
CEC Base Plate	Carbon Steel	1.5 (1)
Inlet Plenum Bottom Plate	Carbon steel	0.5 (1)
Inlet Plenum Shell	Carbon steel	0.5 (1)
Divider Shell	Carbon Steel	0.5 (1)
Divider Shell Flange	Carbon Steel	1 (1)
Closure Lid Strongback	Carbon steel	1
Closure Lid Bottom Plate	Carbon steel	1
Closure Lid Outer Shell	Carbon steel	1 [2]
Closure Lid Cover Plate	Carbon steel	1
Closure Lid Lower Shield Shell	Carbon steel	0.5
Closure Lid Lower Shield Top Plate	Carbon steel	0.5
Closure Lid Lower Shield Bottom Plate	Carbon Steel	1 (1)

Note 1: The thickness of the noted items is not an optional specification. Thicknesses are defined in the HI-STORM UMAX drawings provided in Chapter 1.

Note 2: See Bullet 7 in Section 5.3.

[PROPRIETARY INFORMATION WITHHELD IN ACCORDANCE WITH 10 CFR 2.390]

FIGURE 5.3.1: HI-STORM UMAX MODULE CROSS SECTIONAL ELEVATION VIEW WITH MISSILE PENETRATION. (SKY COLOR REPRESENTS CONCRETE, PINK REPRESENTS SUBGRADE, ORANGE REPRESENTS AIR AND YELLOW REPRESENTS MPC)

[PROPRIETARY INFORMATION WITHHELD IN ACCORDANCE WITH 10 CFR 2.390]

FIGURE 5.3.2: HI-STORM UMAX MODULE CROSS SECTIONAL ELEVATION VIEW WITH DOSE POINT LOCATIONS

5.4 SHIELDING EVALUATION

Shielding analyses methodology applied to HI-STORM UMAX is identical with all the previous Holtec International's licensing applications. The MCNP5 code was used for all of the shielding analyses [5.0.4]. MCNP is a continuous energy, three-dimensional, coupled neutron-photon-electron Monte Carlo transport code. Continuous energy cross section data are represented with sufficient energy points to permit linear-linear interpolation between points. The individual cross section libraries used for each nuclide are those recommended by the MCNP manual. Cross section libraries are based on ENDF/B-V and ENDF/B-VI, except for Sn isotopes where the ENDL92 library is used, and uranium isotopes where LANL/T16 libraries are used. These are the default libraries for the MCNP code version used here [5.0.4]. MCNP has been extensively benchmarked against experimental data by the large user community. References [5.4.1], [5.4.2], and [5.4.3] are three examples of the benchmarking that has been performed.

The energy distribution of the source term, as described earlier, is used explicitly in the MCNP model. A different MCNP calculation is performed for each of the three source terms (neutron, decay gamma, and ^{60}Co). The axial distribution of the fuel source term is described in Table 2.1.5 and Figures 2.1.3 and 2.1.4 of the HI-STORM FW FSAR [5.0.3]. The PWR and BWR axial burn-up distributions were obtained from References [5.4.4] and [5.4.5], respectively, and have previously been utilized in the HI-STORM FSAR [5.0.2]. These axial distributions were obtained from operating plants and are representative of PWR and BWR fuel with burnups greater than 30,000 MWD/MTU. The ^{60}Co source in the hardware was assumed to be uniformly distributed over the appropriate regions.

It has been shown that the neutron source strength varies as the burnup level raised by the power of 4.2. Since this relationship is non-linear and since the burnup in the axial center of a fuel assembly is greater than the average burnup, the neutron source strength in the axial center of the assembly is greater than the relative burnup times the average neutron source strength. In order to account for this effect, the neutron source strength in each of the 10 axial nodes listed in Table 2.1.5 in [5.0.3] was determined by multiplying the average source strength by the relative burnup level raised to the power of 4.2. The peak relative burnups listed in Table 2.1.5, loc. cit., for the PWR fuel is 1.105. By employing the power of 4.2 relationship, the neutron source strength in the peak nodes for the PWR fuel increases by 37.6% ($1.105^{4.2}/1.105$). The total neutron source strength increases by 15.6%.

MCNP was used to calculate doses at the various desired locations. MCNP calculates neutron or photon flux and these values can be converted into dose by the use of dose response functions. This is done internally in MCNP and the dose response functions are listed in the input file in Appendix 5.A. The response functions used in these calculations are listed in Table 5.4.1 and were taken from ANSI/ANS 6.1.1, 1977 [5.4.6].

The dose rates at the various locations were calculated with MCNP using a two-step process. The first step was to calculate the dose rate for each dose location per starting particle for each neutron and gamma group in each basket region for each axial and azimuthal dose location. The

second step is to multiply the dose rate per starting particle for each energy group and basket location (i.e., tally output/quantity) by the source strength (i.e., particles/sec) in that group and sum the resulting dose rates for all groups and basket locations in each dose location. The normalization of these results and calculation of the total dose rate from neutrons, fuel gammas or Co-60 gammas is performed with the following equation.

$$T_{final} = \sum_{j=1}^{M}\left[\sum_{i=1}^{N}\frac{T_{i,j}}{Fm_i} * F_{i,j}\right] \quad \text{(Equation 5.4.1)}$$

where,

T_{final} = Final dose rate (rem/h) from neutrons, fuel gammas, or Co-60
N = Number of groups (neutrons, fuel gammas) or Number of axial sections (Co-60 gammas)
M = Number of regions in the basket
$T_{i,j}$ = Tally quantity from particles originating in MCNP in group/section i and region j (rem/h)(particles/sec)
$F_{i,j}$ = Fuel Assembly source strength in group i and region j (particles/sec)
Fm_i = Source fraction used in MCNP for group i

Note that dividing by Fm_i (normalization) is necessary to account for the number of MCNP particles that actually start in group i. Also note that T_i is already multiplied by a dose conversion factor in MCNP.

The standard deviations of the various results were statistically combined to determine the standard deviation of the total dose in each dose location. The estimated variance of the total dose rate, S^2_{total}, is the sum of the estimated variances of the individual dose rates S^2_i. The estimated total dose rate, estimated variance, and relative error [5.0.4] are derived according to Equations 5.4.2 through 5.4.5.

$$R_i = \frac{\sqrt{S_i^2}}{T_i} \quad \text{(Equation 5.4.2)}$$

$$S_{Total}^2 = \sum_{i=1}^{n} S_i^2 \quad \text{(Equation 5.4.3)}$$

$$T_{Total} = \sum_{i=1}^{n} T_i \quad \text{(Equation 5.4.4)}$$

$$R_{Total} = \frac{\sqrt{S_{Total}^2}}{T_{Total}} = \frac{\sqrt{\sum_{i=1}^{n} S_i^2}}{T_{Total}} = \frac{\sqrt{\sum_{i=1}^{n} (R_i \times T_i)^2}}{T_{Total}} \quad \text{(Equation 5.4.5)}$$

where,

i	=	tally component index
n	=	total number of components
T_{Total}	=	total estimated tally
T_i	=	tally *i* component
S^2_{Total}	=	total estimated variance
S^2_i	=	variance of the *i* component
R_i	=	relative error of the *i* component
R_{Total}	=	total estimated relative error

Note that the two-step approach outlined above allows the accurate consideration of the neutron and gamma source spectrum, and the location of the individual assemblies, since the tallies are calculated in MCNP as a function of the starting energy group and the assembly location, and then in the second step multiplied with the source strength in each group in each location. It is therefore equivalent to a one-step calculation where source terms are directly specified in the MCNP input files, except for the following approximations:

- Fuel is modeled as fresh UO_2 fuel (rather than spent fuel) in MCNP, with an upper bound (5%) enrichment.
- The axial burnup profile is modeled by assigning a source probability to each of the 10 axial sections of the active region, based on a representative axial burnup profile [5.0.2]. For fuel gammas, the probability is proportional to the burnup, since the gamma source strength changes essentially linearly with burnup. For neutrons, the probability is proportional to the burnup raised to the power of 4.2, since the neutron source strength is proportional to the burnup raised to about that power [5.4.7]. This is a standard approach that has been previously used in the licensing calculations for the HI-STAR cask models [5.4.8] and [5.4.9] and HI-STORM overpack models [5.0.2] and [5.0.3].

Tables 5.1.1 and 5.1.2 provide dose rates adjacent to and at 1 meter distance from the HI-STORM UMAX module during normal conditions for the MPC-32 and MPC-37, respectively. Table 5.1.3 provides the annual dose at 100 meters from a HI-STORM UMAX module for the MPC-32 and MPC-37 including the contribution from BPRAs. These results clearly demonstrate that the off-site dose from a HI-STORM UMAX is a small fraction of the off-site dose from the aboveground HI-STORM systems. In addition, Table 5.1.4 presents the dose for the accident condition caused by an impact of tornado borne missile on the radiation protection space boundary.

Soil is used to represent Self-hardening Engineering Subgrade in the models between the modules, with a composition and density shown in Table 5.3.2, representing typical soil conditions [5.4.10]. This is conservative, since the areas between and around the modules would contain engineered fill with a typical density higher than soil. Furthermore, the dose rates around the VVM are dominated by the streaming through the inlet and outlet vents, and not by direct radiation through the soil and concrete. To substantiate this, a complete dose rate profile across the lid and the ISFSI pad was determined. For the ISFSI pad, two conditions were evaluated, the normal condition and a condition where the streaming from the inlets and outlets were artificially blocked. For this second condition, dose rates were also calculated at a distance of 100 m from the VVM. This would indicate what portion of the dose rate results from direct radiation through the concrete and soil of the ISFSI pad as opposed to radiation from the streaming from the air inlet and outlet. The dose locations for the profile are shown in Figure 5.3.2, and are labeled alphabetically (A through X). The calculated dose rates are listed in Tables 5.4.2 and 5.4.3 for MPC-32 and MPC-37, respectively. The following conclusions can be drawn from the results:

- The profile did not reveal any locations with dose rates higher than those shown in Figure 5.1.1 and Tables 5.1.1 and 5.1.2.
- On the ISFSI pad, the dose rates are fairly low compared to other areas.
- At a distance of 100 m, the dose rate from the ISFSI pad surface contributes about 20% of the total dose rate.

It is to be noted that site specific analyses to demonstrate compliance with regulatory requirements should use appropriate site specific soil properties if these are substantially different from the properties used in this chapter.

The highest dose rate is observed for the side of the closure lid just above the inlet plenum zone. To investigate the effect of any streaming from the inlet plenum region, additional calculations were performed for dose locations above the plenum at various radial locations, on the level of dose point 2. Table 5.4.5 presents the calculated dose rates for those locations, as a function of the radial distance from dose location 2, for the MPC-37. The results show that the maximum dose rate occurs at a distance of about 4 feet from the surface point 2. This information should be used by the radiation protection team to ensure operational activities around the lid following ALARA principles.

5.4.1 Excavation Dose Analysis

ISFSIs with HI-STORM UMAX VVMs might be built in one stage or in several stages. If the ISFSI is built in several stages, then excavation work will be necessary in the vicinity of the section of the ISFSI that is already in operation and contains loaded modules. To protect workers from radiation from the loaded modules, a radiation protection space (RPS) boundary is defined around the ISFSI in the drawing in Section 1.5. The RPS boundary is placed so that in a radial direction, a minimum of 10.75 ft of engineered fill remains between the construction site and the closest loaded module. For a loaded module on the periphery of the ISFSI, this places the boundary at least 15 ft from the center of the module. For a loaded module not on the periphery

of the ISFSI, this places the boundary at least 22 feet from the center of the module. Calculations were conservatively performed with the 6.5 ft remaining fill (soil is used in the model) around a loaded VVM instead of the 10.75 ft required by the RPS. The dose rates at the surface of the excavation are presented in table 5.4.4 for both MPC-32 and MPC-37. This dose rate is very low, specifically lower than the dose rates at 1 m from the inlet/outlet vents of the modules. The dose rates at a construction site might therefore be dominated by the dose rates from the inlet/outlet vents, and depending on the loading condition of the operating part of the ISFSI, temporary shielding might be used to reduce dose rates to the construction site. It is to be noted that 6.5 feet of soil is considered for this purpose without any concrete enclosure wall.

Table 5.4.1

FLUX-TO-DOSE CONVERSION FACTORS (FROM [5.4.6])

Gamma Energy (MeV)	**(rem/hr)/ (photon/cm^2-s)**
0.01	3.96E-06
0.03	5.82E-07
0.05	2.90E-07
0.07	2.58E-07
0.1	2.83E-07
0.15	3.79E-07
0.2	5.01E-07
0.25	6.31E-07
0.3	7.59E-07
0.35	8.78E-07
0.4	9.85E-07
0.45	1.08E-06
0.5	1.17E-06
0.55	1.27E-06
0.6	1.36E-06
0.65	1.44E-06
0.7	1.52E-06
0.8	1.68E-06
1.0	1.98E-06
1.4	2.51E-06
1.8	2.99E-06
2.2	3.42E-06

Table 5.4.1 (continued)

FLUX-TO-DOSE CONVERSION FACTORS (FROM [5.4.6])

Gamma Energy (MeV)	**(rem/hr)/ (photon/cm^2-s)**
2.6	3.82E-06
2.8	4.01E-06
3.25	4.41E-06
3.75	4.83E-06
4.25	5.23E-06
4.75	5.60E-06
5.0	5.80E-06
5.25	6.01E-06
5.75	6.37E-06
6.25	6.74E-06
6.75	7.11E-06
7.5	7.66E-06
9.0	8.77E-06
11.0	1.03E-05
13.0	1.18E-05
15.0	1.33E-05

Table 5.4.1 (continued)

FLUX-TO-DOSE CONVERSION FACTORS
(FROM [5.4.6])

Neutron Energy (MeV)	Quality Factor	(rem/hr)†/(n/cm^2-s)
2.5E-8	2.0	3.67E-6
1.0E-7	2.0	3.67E-6
1.0E-6	2.0	4.46E-6
1.0E-5	2.0	4.54E-6
1.0E-4	2.0	4.18E-6
1.0E-3	2.0	3.76E-6
1.0E-2	2.5	3.56E-6
0.1	7.5	2.17E-5
0.5	11.0	9.26E-5
1.0	11.0	1.32E-4
2.5	9.0	1.25E-4
5.0	8.0	1.56E-4
7.0	7.0	1.47E-4
10.0	6.5	1.47E-4
14.0	7.5	2.08E-4
20.0	8.0	2.27E-4

† Includes the Quality Factor.

Table 5.4.2

DOSE RATES ADJACENT TO THE HI-STORM UMAX MODULE WITH MPC-32 AT DOSE LOCATIONS SHOWN IN FIGURE 5.3.2

Dose Location††	Dose Rate (mrem/hr unless noted)		Dose Location	Dose Rate (mrem/hr unless noted)	
	Inlet/Outlet Open	Inlet/Outlet Artificially Closed		Inlet/Outlet Open	Inlet/Outlet Artificially Closed
A	2.55	0.11	N	4.61	0.78
B	1.91	0.11	O	8.56	1.90
C	1.25	0.32	P	19.05	5.98
D	1.49	0.28	Q	66.04	16.96
E	1.67	0.32	R	1.44	0.31
F	2.06	0.24	S	1.29	0.28
G	3.10	0.23	T	1.18	0.28
H	0.73	0.32	U	1.10	0.27
I	0.36	0.16	V	1.03	0.26
J	0.59	0.31	W	0.88	0.26
K	0.92	0.49	X	0.80	0.23
L	0.52	0.37	Y	8.58 (mRem/year)	1.58 (mRem/year)
M	0.47	0.22			

†† See Figure 5.3.2

Table 5.4.3

DOSE RATES ADJACENT TO THE HI-STORM UMAX MODULE WITH MPC-37 AT DOSE LOCATIONS SHOWN IN FIGURE 5.3.2

Dose Location††	Dose Rate (mrem/hr unless noted)		Dose Location	Dose Rate (mrem/hr unless noted)	
	Inlet/Outlet Open	Inlet/Outlet Artificially Closed		Inlet/Outlet Open	Inlet/Outlet Artificially Closed
A	2.08	0.10	N	3.08	0.62
B	1.52	0.09	O	5.24	1.50
C	1.02	0.27	P	12.13	4.82
D	1.15	0.24	Q	41.47	13.73
E	1.42	0.22	R	1.01	0.26
F	1.73	0.21	S	0.89	0.25
G	2.35	0.20	T	0.78	0.24
H	0.57	0.26	U	0.74	0.24
I	0.29	0.11	V	0.66	0.22
J	0.51	0.27	W	0.63	0.20
K	0.85	0.53	X	0.55	0.21
L	0.38	0.32	Y	5.96 mrem/year	1.31 (mrem/year)
M	0.34	0.18			

†† See Figure 5.3.2

Table 5.4.4 MAXIMUM DOSE RATES ON THE SURFACE OF THE RADIATION PROTECTION SPACE BOUNDARY	
MPC-32 (mrem/hr)	MPC-37 (mrem/hr)
0.062	0.065

Table 5.4.5

DOSE RATE VALUES FOR LOCATIONS AT RADIAL DISTANCES FROM SURFACE LOCATION 2 FOR HI-STORM UMAX WITH MPC-37

Radial Distance from Surface Location 2 (cm)	Neutrons (mrem/hr)	Photons[§§§] (mrem/hr)	Total (mrem/hr)
61	0.12	0.32	0.44
100	1.53	3.39	4.92
130	2.55	7.09	9.64
161	1.47	4.28	5.75
191	0.95	3.15	4.10
222	0.63	2.51	3.14
252	0.45	2.15	2.60
283	0.33	1.64	1.97

[§§§] Photon dose rates include dose rates from fuel gammas, n-gamma reactions and Co-60 sources. In addition, BPRAs are included in the photon dose rate.

5.5 REGULATORY COMPLIANCE

Chapters 1, 2 and this chapter of this FSAR describe in detail the shielding structures, systems, and components (SSCs) important-to-safety.

The shielding-significant SSCs important-to-safety have been evaluated in this chapter and their impact on personnel and public health and safety resulting from operation of an independent spent fuel storage installation (ISFSI) utilizing the HI-STORM UMAX System has also been evaluated.

In summary it can be concluded that the shielding of the HI-STORM UMAX System is in compliance with 10CFR72 and satisfies the applicable design and acceptance criteria including 10CFR20. Thus, this shielding evaluation provides reasonable assurance that the HI-STORM UMAX system will allow safe storage of spent fuel in full conformance with 10CFR72.

5.6 REFERENCES

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APPENDIX 5.A

SAMPLE INPUT FILE FOR MCNP

[PROPRIETARY INFORMATION WITHHELD IN ACCORDANCE WITH 10 CFR 2.390]

CHAPTER 6: CRITICALITY EVALUATION

6.0 INTRODUCTION

This chapter discusses the criticality safety of the HI-STORM UMAX system. Criticality safety depends foremost on the MPC and fuel basket, and the MPC content. Those are identical between the HI-STORM UMAX and the HI-STORM FW. Criticality safety of the HI-STORM UMAX is therefore demonstrated in this chapter by reference to criticality evaluations documented in Chapter 6 of the HI-STORM FW FSAR [2.0.1], with appropriate recognition of the differences in the overpack configuration. As noted in other chapters, Revision 3 of the HI-STORM FW FSAR has been added to this docket to eliminate the need to consult the HI-STORM FW FSAR in another docket.

6.1 ACCEPTANCE CRITERIA

The acceptance criteria for criticality evaluations for the HI-STORM UMAX system are presented in Subsection 2.0 of this FSAR.

6.2 EVALUATION

During storage conditions in the HI-STORM UMAX system, the maximum k_{eff} will be significantly below the limiting maximum k_{eff} since the MPC is internally dry. Under this condition, the configuration is very similar in all HI-STORM models, which consists of an internally dry MPC, an air gap between the MPC and the overpack, a steel shell or shells and concrete (above-ground) or soil (underground). Results for the HI-STORM UMAX VVM would therefore be practically identical to the results listed for storage conditions in Chapter 6 of the HI-STORM FW FSAR [2.0.1]. Any small differences in results would not affect the principal conclusions, since the maximum k_{eff} under storage conditions (dry inert environment) is substantially below the regulatory limit. Note that the analysis for the MPCs in the HI-STORM documented in Chapter 6 of the HI-STORM FW FSAR [2.0.1] conservatively assume that the gap between the MPC and the HI-STORM is flooded with water, thus increasing the neutron reflection compared to a dry cavity [2.0.1, Section 6.1]. Flooding under accident conditions of the UMAX is therefore also covered by the calculations for the HI-STORM FW. All other normal, off-normal and accident conditions in the HI-STORM UMAX system are identical to those in the HI-STORM FW, since the MPCs, including content, and HI-TRAC are identical between the two systems.

In summary, the limiting condition for storage of MPCs in HI-STORM UMAX is identical to the limiting conditions for the HI-STORM FW from a criticality perspective, and all other normal, off-normal and accident conditions are identical or equivalent between the systems from a criticality perspective. All results and conclusions for criticality safety of the MPCs previously analyzed in Chapter 6 of the HI-STORM FW FSAR [2.0.1] remain applicable to the HI-STORM UMAX system, and no additional calculations to demonstrate criticality safety are required for the HI-STORM UMAX system.

CHAPTER 7: CONFINEMENT EVALUATION

7.0 INTRODUCTION

This chapter discusses the confinement safety of the HI-STORM UMAX System. Confinement of all radioactive materials in the HI-STORM UMAX system is provided by the MPC. MPCs are identical between the HI-STORM UMAX and the HI-STORM FW. All normal, off-normal and accident conditions relevant for confinement are identical between the HI-STORM UMAX and the HI-STORM FW, and there are no new conditions for the HI-STORM UMAX system that would require additional confinement analyses. Confinement safety of the HI-STORM UMAX is therefore demonstrated in this chapter by reference to confinement evaluations documented in Chapter 7 of the HI-STORM FW [2.0.1], with appropriate recognition of the differences in the overpack configuration. For ease of reference, the entire body of Revision 3 of the HI-STORM FW FSAR has been placed in this docket.

7.1 ACCEPTANCE CRITERIA

The acceptance criteria for confinement evaluations for the HI-STORM UMAX system are presented in Chapter 2 of this FSAR.

7.2 EVALUATION

The MPCs will be stored in the HI-STORM UMAX VVM in a passive state just as they are stored in the above ground overpacks described in the HI-STORM FW FSAR. Furthermore, as shown in Chapter 4, the temperature field in the MPC is engineered to be bounded by that determined in the HI-STORM FW FSAR. Therefore, the stress levels in the MPC pressure retaining boundary will be bounded by that in its certification basis value in the HI-STORM FW FSAR which leads to the axiomatic conclusion that the confinement integrity determinations reached in the HI-STORM FW MPC's also apply to storage in HI-STORM UMAX.

In summary, the storage configuration of MPCs in HI-STORM UMAX is identical to the storage configuration in the HI-STORM FW from a confinement perspective. Therefore, all descriptions and conclusions for the confinement system presented in Chapter 7 of the HI-STORM FW [2.0.1] remain applicable to the HI-STORM UMAX system, and no additional confinement evaluations are required for the HI-STORM UMAX system.

CHAPTER 8: MATERIAL EVALUATION

8.1 INTRODUCTION

This chapter presents an assessment of the materials selected for use in the HI-STORM UMAX system components identified in the licensing drawings in Section 1.5. The assessment of the materials selected for use in the MPC and HI-TRAC (i.e., components common to HI-STORM UMAX system and the previously licensed HI-STORM FW system) is provided in Chapter 8 of the HI-STORM FW FSAR. Material considerations pertaining to the components common to the "FW" and the "UMAX" systems, are referenced in this FSAR, as appropriate, to the HI-STORM FW FSAR. To facilitate convenient access to the referenced material, the latest edition of the HI-STORM FW FSAR has been placed in this docket and a list of “FW” FSAR sections germane to this chapter is provided in the (unnumbered) table below.

APPLICABLE SECTIONS OF HI-STORM FW FSAR		
Location of UMAX FSAR	**Subject of the Reference**	**Location in HI-STORM FW FSAR, Revision 3***
Section 8.1	Material evaluation for use in the MPCs and HI-TRAC	Paragraph 8.2.1.2, 8.2.1.2.i and 8.2.1.2.iii
Table 8.1.1	Performance of materials used in the MPC and HI-TRAC for short term operations	Table 8.1.1 and 8.1.3
Table 8.1.4	Failure and degradation mechanisms related to the performance of materials used in the MPCs and HI-TRAC for short term and storage operations	Table 8.1.4
Paragraph 8.2.1.2.iv	Material selection evaluation of austenitic stainless steel	Paragraph 8.2.1.2.i
Section 8.4	Material Properties for MPCs and HI-TRAC	Paragraph 8.4.4.1
Section 8.5	Welding material and welding specification for MPC and HI-TRAC	Section 8.5
Section 8.7	Coatings and corrosion mitigation techniques for MPC and HI-TRAC	Sub-Section 8.7.2 and 8.7.3
Section 8.8	Gamma and neutron shielding materials used in HI-TRAC	Section 8.8
Section 8.9	Neutron Absorbing Materials	Section 8.9
Section 8.10	Seals for the HI-TRAC bottom lid	Section 8.11

* For convenience of reference, the specific revision of the HI-STORM FW FSAR that is referenced in the safety analysis herein is placed in this docket. Updated versions of the HI-STORM FW FSAR shall be placed in this docket as necessary so as to ensure that the safety analyses on the "UMAX" docket (72-1040) remain aligned with the material referenced in the HI-STORM FW FSAR.

Section 8.11	Chemical and galvanic reactions related to the MPC and HI-TRAC	Section 8.12
Section 8.12	Fuel cladding integrity during short term operations	Section 8.13
Section 8.13	Examination and testing requirements for the MPC and HI-TRAC	Section 8.14.1

In this chapter and in Chapters 2 and 3 of this FSAR, the significant mechanical, thermal, radiological, and metallurgical properties of materials identified for use in the components of the HI-STORM UMAX System VVM and ISFSI are presented.

The HI-STORM UMAX components must withstand the environmental conditions experienced during normal operation, off-normal conditions, and accident conditions for the entire service life. The major structural materials used in HI-STORM UMAX System are discussed in this chapter.

Chapter 1 provides a general description of the HI-STORM UMAX System including information on materials of construction. The ITS categories of the principal materials of construction in the HI-STORM UMAX VVM and ISFSI system are identified in the drawing package provided in Section 1.5.

The Materials selected for the HI-STORM UMAX VVM are the same as those used in its anatomically similar predecessor, HI-STORM 100U VVM, licensed in Docket number 72-1014, except the subgrade between the ISFSI pad and the Support Foundation Pad (SFP) which is labeled as Space A in Figure 2.4.4 is restricted to CLSM or lean concrete. Therefore, the material considerations for HI-STORM UMAX are substantially parallel to those for HI-STORM 100U.

Nevertheless, for completeness, it is necessary that the material considerations applicable to HI-STORM UMAX be independently evaluated for compliance with the ISG-15 [8.1.1] which contains the latest NRC position in this matter. The principal purpose of ISG-15 is to evaluate the dry cask storage system to ensure adequate material performance of components deemed to be important-to-safety at an independent spent fuel storage installation (ISFSI) under normal, off-normal, and accident conditions. Guidance on performing the safety evaluation of the materials is adopted directly from ISG-15.

ISG-15 sets down the following general acceptance criteria for material evaluation:

- The safety analysis report should describe all materials used for dry spent fuel storage components important-to-safety, and should consider the suitability of those materials for their intended functions in sufficient detail to evaluate their effectiveness in relation to all safety functions.

- The dry spent fuel storage system should employ materials that are compatible with wet and dry spent fuel loading and unloading operations and facilities. These materials should not degrade to the extent that a safety concern is created.

The information compiled in this chapter seeks to address the above acceptance criteria in full measure for the HI-STORM UMAX VVM and ISFSI. To perform the material suitability evaluation, it is necessary to characterize the following for each component: (i) the applicable environment, (ii) potential degradation modes and (iii) potential hazards to continued effectiveness of the selected material.

This FSAR seeks to qualify previously licensed MPCs (Table 1.2.1) in the HI-STORM FW docket for long-term storage in the HI-STORM UMAX modules The information required to address the short-term operations for the MPC and HI-TRAC are contained in Chapter 10 of the HI-STORM FW FSAR. The material evaluation effort, therefore, is directed towards the long-term storage and its consequences to the system's continued safety. Tables 8.1.1 and 8.1.2 provide a summary of the environmental states, potential degradation modes, and hazards applicable to the HI-STORM UMAX modules. Table 2.6.2 provides a listing of permissible materials used in the HI-STORM UMAX system. Table 8.1.3 provides the listing of material types that are important to safety and are subject to the ambient environmental of an ISFSI.

To provide a proper context for the subsequent evaluations, the potential degradation mechanisms applicable to the ventilated systems are summarized in Table 8.1.4. The degradation mechanisms listed in Table 8.1.4 are considered in the suitability evaluation presented in this chapter.

The material evaluation presented in this chapter is intended to be complete, even though *a' priori* conclusion of the adequacy of the materials can be made on the basis of the following facts:

i. The materials used in HI-STORM UMAX VVM are identical to those used in the widely deployed HI-STORM 100 System (Docket No. 72-1014) [8.1.2] including its underground VVM denoted as HI-STORM 100U and the HI-STORM FW system (Docket number 72-1032) [8.1.3].

ii. The thermal environment in the HI-STORM UMAX system emulates other HI-STORM models in all respects.

iii. The MPC transfer operations, described in Chapter 9 herein, are identical to those that have been practiced in the HI-STORM 100 system throughout the industry.

The organization of technical information in this chapter mirrors the format and content of Chapter 8 in the HI-STORM FW FSAR and complies with the guidance in NUREG-1536 [8.1.4]

Table 8.1.1

CONSIDERATIONS GERMANE TO PERFORMANCE OF MATERIALS USED IN THE MPCs IN LONG TERM STORAGE IN HI-STORM UMAX

Consideration	**Environment**
Environment	MPC's internal environment is hot (≤ 752°F), inertized and dry. Temperature of the MPC internals cycles vary gradually due to changes in the environmental temperature.
Potential degradation modes	Corrosion of the external surfaces of the MPC (stress, corrosion, cracking, pitting, etc.).
Potential hazards to effective performance	Blockage of ventilation ducts under an extreme environmental phenomenon leading to a rapid heat-up of the MPC internals.

Note that the considerations germane to the performance of materials used in the MPC and HI-TRAC for short term operations are addressed in Section 8.1 of the HI-STORM FW FSAR.

Table 8.1.2

CONSIDERATIONS GERMANE TO THE HI-STORM UMAX VVM MATERIAL PERFORMANCE

Consideration	**Performance Data**
Environment	Cool ambient air is progressively (but marginally) heated as it flows up the annulus between the Divider Shell and the MPC heating the inside surface of the cask and cooling the outside surface of the MPC. The heated air has reduced relative humidity the warmer it gets. As a result, the bottom external surface of the Closure Lid is heated and the top external surfaces are in contact with ambient air, rain, and snow, as applicable. The exterior surfaces of the CEC are in contact with either engineered fill or concrete (concrete encasement or "free-flow "concrete) and may be subjected to cathodic protection, as applicable.
Potential degradation modes	Peeling or perforation of surface preservatives on steel surfaces and corrosion of exposed steel surfaces.
Potential hazards to effective performance	Blockage of ducts by debris leading to overheating of the concrete in the overpack, scorching of the cask by proximate fire, lightning.

Table 8.1.3
MATERIAL TYPES IN THE HI-STORM UMAX SYSTEM COMPONENTS EXPOSED TO THE LONG-TERM AMBIENT ENVIRONMENT

	Material Type	**Components and Their Surfaces Exposed to Ambient Environment**
1.	Low carbon steel	• All surfaces of the closure lid • Internal surfaces of the CEC (expose to air) • External surfaces of the CEC (exposed to buffer concrete) or subgrade • Internal and External surfaces of the Divider shell
2.	Shielding concrete	• The outside surface of the ISFSI pad
3.	Alloy X Austenitic Stainless Steel	• External surfaces of the stored MPC • MPC Guides and MPC support surfaces inside the CEC. • Surfaces of the closure lid (optional per Section 1.5) • Internal surfaces of the CEC (optional per Section 1.5) • External surfaces of the CEC (optional per Section 1.5) • Internal External surfaces of the Divider shell (optional per Section 1.5)
4.	Elastomeric Gasket	• Closure Lid Seal • Divider Shell Seal

Table 8.1.4

FAILURE AND DEGRADATION MECHANISMS*

	Mechanism	Area of Performance Affected	Vulnerable Parts
1.	General Corrosion	Structural capacity	All carbon steel parts
2.	Stress Corrosion Cracking	Structural	Austenitic Stainless Steel
3.	Galling	Equipment handling and deployment	Threaded Fasteners
4.	Fatigue	Structural Integrity	Fuel Cladding & Bolting
5.	Brittle Fracture	Structural Capacity	Thick Steel Parts
6.	Boron Depletion	Criticality Control	Neutron Absorber

Note that the failure and degradation mechanisms related to the performance of materials used in the MPC and HI-TRAC for short term and storage operations are addressed in Section 8.1 of the HI-STORM FW FSAR.

* This table lists all potential (generic) mechanisms, whether they are credible for the HI-STORM UMAX System or not. The viability of each failure mechanism is discussed later in this chapter.

8.2 MATERIAL SELECTION

The acceptance criteria for the materials subject to long-term storage conditions in HI-STORM UMAX are extracted from ISG-15[8.1.1] as follows:

a. The material properties of a dry spent fuel storage component should meet its service requirements in the proposed cask system for the duration of the licensing period.

b. The materials that comprise the dry spent fuel storage should maintain their physical and mechanical properties during all conditions of operations. The spent fuel should be readily retrievable without posing operational safety problems.

c. Over the range of temperatures expected prior to and during the storage period, any ductile-to-brittle transition of the dry spent fuel storage materials, used for structural and nonstructural components, should be evaluated for its effects on safety.

d. Dry spent fuel storage gamma shielding materials should not experience slumping or loss of shielding effectiveness to an extent that compromises safety. The shield should perform its intended function throughout the licensed service period.

e. Dry spent fuel storage materials used for neutron absorption should be designed to perform their safety function.

f. Dry spent fuel storage protective coatings should remain intact and adherent during all loading and unloading operations within wet or dry spent fuel facilities, and during long-term storage.

It is recognized that the qualification of the materials used in the MPC types and the HI-TRAC transfer cask is documented in Section 8.2 of the HI-STORM FW FSAR. The material selection opportunities for the HI-STORM UMAX system, therefore, are limited to the VVM module assembly components and the reinforced concrete structures that support or surround them.

However, to obviate any new material qualification effort, the materials permitted for the HI-STORM UMAX system *are limited to those certified in other HI-STORM 100 and HI-STORM FW dockets*. The material qualification information presented in this chapter is accordingly adopted from Docket Number 72-1032.

8.2.1 Structural Materials

8.2.1.1 Cask Components and Their Constituent Materials

The major structural material that is used in the HI-STORM UMAX VVM is steel. The concrete in the VVM Closure Lid does not play a major structural role but is present in large quantity for the main purpose of shielding. The major structural materials in the ISFSI structures are the concrete and rebars in the Support Foundation Pad, the ISFSI Pad, and the Self-hardening Engineered Subgrade (SES).

8.2.1.2 Synopsis of Structural Materials

i. Carbon Steel, Low-Alloy Steel

Materials for the HI-STORM UMAX VVM are selected to preclude brittle fracture. Details of discussions are provided in Section 3.1 and Section 8.4, as applicable. The fracture toughness test requirements specified in Table 3.1.9 apply to the HI-STORM UMAX Closure Lid ferritic structural steels including any "significant-to-handling" (STH) parts of the Closure Lid.

ii. Reinforced Concrete

All reinforced concrete load bearing structures (concrete and rebar) in the HI-STORM UMAX ISFSI will conform to stress criteria of ACI-318(2005) [8.2.1]. Section 3.3 provides properties for reinforced concrete to be used for the HI-STORM UMAX interfacing ISFSI structures. The service life of the ISFSI structures is specified to be the same as that of the HI-STORM UMAX VVM.

iii. Self-hardening Engineered Subgrade

The SES material (i.e., lean concrete or CLSM) used in the HI-STORM UMAX ISFSI will conform to the stress criteria of ACI-318(2005) or ACI-229(1999). Tables 2.3.2 and 3.3.4 provide the critical properties for the SES material used for HI-STORM UMAX ISFSI safety analyses. In the interest of a reliably robust design and long service life, additional performance properties of CLSM are listed in table below. The service life for the SES is the same as that of the VVM and ISFSI reinforced concrete.

iv. Austenitic Stainless Steel

Austenitic stainless steel may be used for certain components of the HI-STORM UMAX VVM. Chapter 3 provides the structural evaluation for the HI-STORM UMAX VVM using the governing structural materials. Since stainless steel materials do not undergo a ductile-to-brittle transition in the minimum permissible service temperature range of the HI-STORM UMAX System, brittle fracture is not a concern for stainless steel components. It is

recognized that austenitic stainless steels are qualified for use with other HI-STORM UMAX System components (namely Alloy X for the MPC) by the HI-STORM FW FSAR.

Additional CLSM Performance Properties*		
Performance Property	**Test Property**	**Nominal Value**
Corrosive Resistance	pH Resistivity Permeability	7.5 – 11.5 > 279000 ohm-cm < 10^{-5} cm/sec
Flowability	Flow	6” – 8” (ASTM D 6103)
Excavatability	Unconfined Compressive Strength	Not excavatable since compressive strength is greater than 300 psi
Permeability	Water Permeability	< 10^{-5} cm/sec
Strength	Penetration Resistance	> 650
Acidity/Alkalinity	pH	7.5 – 11.5
Note: * These properties are not used in HI-STORM UMAX safety analyses; nominal values obtained from References [8.2.3], [8.2.4], and [8.2.5] are tabulated for information only.		

Chapter 3 discusses the structural evaluations of the HI-STORM UMAX System components and ISFSI structures. It is demonstrated that the structural steel components of the HI-STORM UMAX VVM and the SFP concrete meet the allowable stress limits for normal, off-normal, and accident loading conditions as applicable. The analyses documented in Chapter 3 also demonstrate that the SES remains stable under the Design Basis Earthquake condition and provides sufficient protection to the stored MPC even if any side of the self hardening sub-grade (SES) is fully exposed during excavation for ISFSI expansion.

8.2.2 Non-Structural Materials

i. Plain Concrete

Plain concrete is specified for the VVM Closure Lid for its shielding properties and also as an encasement around the exterior of the VVM CEC shell, if required, for its corrosion mitigation properties. Lean concrete (plain concrete with a lower cement ratio and hence lower compressive strength) is also specified as a suitable material for the structural subgrade below the ISFSI pad. The requirements on the shielding concrete are specified in Table 2.3.2 in Chapter 2.

The shielding performance of the plain concrete is maintained by ensuring that the minimum concrete density is met during construction and the allowable concrete temperature limits are not exceeded. The thermal analyses for normal and off-normal conditions are carried out in this FSAR to insure that the plain concrete does not exceed the allowable long term temperature limit provided in Table 2.3.7.

ii. Insulation

The Divider Shell is lined with insulation on its outer surface to prevent excessive heating of the ISFSI pad. The insulation selected shall be suitable for high temperature and high humidity operation and shall be foil faced, jacketed, or otherwise made water-resistant to ensure the required thermal resistance is maintained in accordance with Chapter 4. The high zinc content present in the coating of the Divider Shell provides protection for the jacketing or foil from the potential of galvanic corrosion. To ensure adequate radiation resistance, the insulation blanket does not contain any organic binders. The damage threshold for ceramics is known to be approximately $1x10^{10}$ Rads. Chloride corrosion is not a concern since chloride leachables are limited and sufficiently low. Stress corrosion cracking of the foil or jacketing, whether made from stainless steel or other material, is not an applicable corrosion mechanism due to minimal stresses derived from self-weight. The foil or jacketing and attachment hardware shall either have sufficient corrosion resistance (e.g., stainless steel, aluminum, or galvanized steel) or shall be protected with a suitable surface preservative. The insulation is adequately secured to prevent blockage of the ventilation passages in case of failure of a single attachment (strap, clamp, bolt or other attachment hardware). The following table provides the acceptance criteria for the selection of insulation material for the VVM assembly and ranks them in order of importance.

Acceptance Criteria for the Selection of the Insulation Material	
Rank	**Criteria**
1	Adequate thermal resistance (See Table 4.1.1)
2	Adequate high temperature resistance (See Table 2.3.7)
3	Adequate humidity resistance
4	Adequate radiation resistance
5	Adequate resistance to the ambient environment
6	Sufficiently low chloride leachables
7	Adequate integrity and resistance to degradation and corrosion during long-term storage

Kaowool® ceramic fiber insulation [8.2.2] is selected as one that satisfies the acceptance criteria to the maximum degree. The Kaowool® insulation material provides excellent resistance to chemical attack and is not degraded by oil or water. Alternatively, a Holtec-approved equivalent that meets the above acceptance criteria may be used.

8.2.3 Critical Characteristics and Equivalent Materials*

As defined in the Glossary, the *critical characteristics* of a material are the properties that the material must possess to enable the part in which it is used to render its intended design function. However, material designations adopted by the International Standards Organization (ISO) also affect the type of steels and steel alloys available from suppliers around the world. Therefore, it is necessary to provide for the ability in this FSAR to substitute materials with equivalent materials in the manufacture of the equipment governed by this FSAR.

As defined in this FSAR, the term “Equivalent Material” has a specific meaning: Equivalent materials are those that can be substituted for each other without adversely affecting the safety function of the SSC (system, structure, and component) in which the substitution is made. Substitution by an equivalent material can be made after the equivalence in accordance with the provisions of this FSAR has been established.

The concept of equivalent materials explained above has been previously used in the HI-STORM 100 FSAR [8.1.2] to qualify four different austenitic stainless steel alloys (ASME SA240 Types 304, 304LN, 316, and 316LN) to serve as candidate MPC materials.

The equivalence of materials is directly tied to the notion of *critical characteristics*. A critical characteristic of a material is a material property whose value must be specified and controlled to ensure an SSC will render its intended function. The numerical value of the critical characteristic invariably enters in the safety evaluation of an SSC and therefore its range must be guaranteed. To ensure that the safety calculation is not adversely affected, properties such as Yield Strength, Ultimate Strength, and Elongation must be specified as minimum guaranteed values. However, there are certain properties where both minimum and maximum acceptable values are required (in this category lies specific gravity and thermal expansion coefficients).

Table 8.2.1 lists the array of properties typically required in safety evaluation of an SSC in dry storage and transport applications. The required value of each applicable property, guided by the safety evaluation needs, defines the critical characteristics of the material. The subset of applicable properties for a material depends on the role played by the material. The role of a material in the SSC is divided into three categories:

* Materials in this section have been adapted from HI-STORM FW FSAR and therefore in Arial font [8.1.3]

Type	Technical Area of Applicability
S	Those needed to ensure structural compliance
T	Those needed to ensure thermal compliance (temperature limits)
R	Those needed to ensure radiation compliance (criticality and shielding)

The properties listed in Table 8.2.1 are the ones that may apply in a dry storage or transport application.

The following procedure shall be used to establish acceptable equivalent materials for a particular application:

Criterion i: Functional Adequacy:
Evaluate the guaranteed critical characteristics of the equivalent material against the values required to be used in safety evaluations. The required values of each critical characteristic must be met by the minimum (or maximum) guaranteed values (MGVs of the selected material).

Criterion ii: Chemical and Environmental Compliance:
Perform the necessary evaluations and analyes to ensure the candidate material will not excessively corrode or otherwise degrade in the operating environment.

A material from another designation regime that meets Criteria (i) and (ii) above is deemed to be an acceptable material, and hence, equivalent to the candidate material. For ITS materials used with the HI-STORM UMAX VVM, recourse to equivalent materials shall be made only if the designated material in this FSAR is not readily available or the use of another equivalent material is deemed to produce a desirable outcome such as improvement in fabricability, absence of lamination, tolerance adherence, etc. An example of an acceptable substitution is replacing carbon steel plate stock (A515 Gr. 70 or 516 Gr. 70) with forging (A105 or A350 LF-2).

As can be ascertained from its definition in the glossary, the *critical characteristics* of the material used in a subcomponent depend on its function. The Closure Lid, for example, serves as a shielding device and as a physical barrier to protect the MPC against loadings under all service conditions, including the Extreme Environmental phenomena. Therefore, the critical characteristics of steel used in the lid are its strength (yield and ultimate), ductility, and fracture resistance.

The appropriate critical characteristics for structural components of the HI-STORM UMAX System, therefore, are:

- Material yield strength, σ_y
- Material ultimate strength, σ_u
- Elongation, ε

- Charpy impact strength at the lowest service temperature for the part, C_i (Unless exempted by other provisions in the governing code)

Thus, the carbon steel specified in the drawing package can be substituted with different steel so long as each of the four above properties in the replacement material is equal to or greater than their minimum values used in the qualifying analyses used in this FSAR.

In the event that one or more of the critical characteristics of the replacement material is slightly lower than the original material, then the use of the §72.48 process shall be necessary to ensure that all regulaory predicates for the material substitution are fully satisfied.

Table 8.2.1

TYPICAL CRITICAL CHARACTERISTICS OF MATERIALS REQUIRED FOR SAFETY EVALUATION OF STORAGE AND TRANSPORT SYSTEMS

	Property	**Type**	**Purpose**	**Bounding Acceptable Value**
1.	Minimum Yield Strength	S	To ensure adequate elastic strength for normal service conditions	Min.
2.	Minimum Tensile Strength	S	To ensure material integrity under accident conditions	Min.
3.	Young's Modulus	S	For input in structural analysis model	Min.
4.	Minimum elongation of δ_{min}, %	S	To ensure adequate material ductility	Min.
5.	Impact Resistance at Ambient Conditions	S	To ensure protection against crack propagation	Min.
6.	Maximum Allowable Creep Rate	S	To prevent excessive deformation under steady state loading at elevated temperatures	Max.
7.	Thermal conductivity (minimum averaged value in the range of ambient to maximum service temperature, t_{max})	T	To ensure that the basket will conduct heat at the rate assumed in its thermal model	Min.
8.	Minimim Emmissivity	T	To ensure that the thermal calculations are performed conservatively.	Min.
9.	Specific Gravity	S (and R)	To compute weight of the component (and shielding effectiveness)	Max. (and Min.)
10.	Thermal Expansion Coefficient	T (and S)	To compute the change in basket dimension due to temperature (and thermal stresses)	Min. (and Max.)
11.	Boron-10 Content	R	To control reactivity	Min.

8.3 APPLICABLE CODES AND STANDARDS

The principle codes and standards applied to the HI-STORM UMAX System components are the ASME Code Section II [8.3.1], the ACI code [8.2.1], the ASTM Standards, and the ANSI standards. Chapter 1 provides details of the specific applications of these codes and standards along with the other codes and standards that are applicable. The principle codes and standards applied to the MPC and HI-TRAC are described in Section 8.3 of the HI-STORM FW FSAR and are not repeated here.

Chapter 2 discusses factored load combinations for ISFSI pad design per NUREG-1536 [8.1.4]. Codes ACI 360R-92, "Design of Slabs on Grade"; ACI 302.1R, "Guide for Concrete Floor and Slab Construction"; and ACI 224R-90, "Control of Cracking in Concrete Structures" are also used in the design and construction of the concrete pad, as appropriate. Section 2.2 elaborates on the specific applications of the ASME Boiler and Pressure Vessel code for the HI-STORM UMAX System.

Chapter 3 provides allowable stresses and stress intensities for various materials extracted from applicable ASME code sections for various service conditions. This chapter also provides discussions on fracture toughness test requirements per ASME code sections. Mechanical properties of materials are extracted from applicable ASME sections [8.3.1], [8.3.2] and are tabulated for various materials used in HI-STORM UMAX System. Concrete properties are from ACI 318-2005 code.

In order to meet the requirements of the codes and standards the materials must conform to the minimum acceptable physical strengths and chemical compositions and the fabrication procedures must satisfy the prescribed requirements of the applicable codes.

Additional codes and standards applicable to welding are discussed in Section 8.5 and those for the bolts and fasteners are discussed in Section 8.6.

Review of the above shows that the identified codes and standards are appropriate for the material control of major components. Additional material control is identified in material specifications. Material selections are appropriate for environmental conditions to be encountered during loading, unloading, transfer, and storage operations. The materials and fabrication of major components are suitable based on the applicable codes of record.

8.4 MATERIAL PROPERTIES

This section provides discussions on material properties that mainly include mechanical and thermal properties. Additional discussions related to materials used in the MPC and HI-TRAC are presented in Section 8.4 of the HI-STORM FW FSAR. The material properties used in the design and analysis of the HI-STORM UMAX System are obtained from established industry sources such as the ASME Boiler and Pressure Vessel Code [8.3.1], ASTM publications, handbooks, textbooks, other NRC-reviewed SARs, and government publications, as appropriate.

8.4.1 Mechanical Properties

Section 3.3 presents mechanical properties of all ITS materials used in the HI-STORM UMAX System. The structural materials include Alloy X, carbon steel, low-alloy and nickel-alloy steel, bolting materials, and weld materials. The properties include yield stress, mean coefficient of thermal expansion, ultimate stress, and Young's modulus of these materials and their variations with temperature. Certain mechanical properties are also provided for nonstructural materials such as concrete used for shielding.

The discussion on mechanical properties of materials in Chapter 3 provides reasonable assurance that the class and grade of the structural materials are acceptable under the applicable construction code of record. Selected parameters such as the temperature dependent values of stress allowables, modulus of elasticity, Poisson's ratio, density, thermal conductivity, and thermal expansion have been appropriately defined in conjunction with other disciplines. The material properties of all code materials are guaranteed by procuring materials from Holtec-approved vendors through the so-called "material dedication" process*, if necessary.

8.4.2 Thermal Properties

Section 4.2 presents thermal properties of materials used in the MPC such as Alloy X, Metamic-HT, aluminum shims and helium gas; materials present in HI-STORM UMAX such as carbon steel, stainless steel and concrete; and materials present in HI-TRAC transfer cask that include carbon steel, lead and demineralized water. The properties include density, thermal conductivity, heat capacity, viscosity, and surface emissivity/absorptivity. Variations of these properties with temperature are also provided in tabular forms.

The thermal properties of fuel (UO_2) and fuel cladding are also reported in Section 4.2. Thermal properties are obtained from standard handbooks or established text books (see Table 4.2.1).

* Dedication is a term of art in nuclear quality assurance.

8.4.3 Low Temperature Ductility of Ferritic Steels*

The risk of brittle fracture in the HI-STORM UMAX components is eliminated by utilizing materials that maintain high fracture toughness under "cold" conditions (temperatures < -40 degrees F)

The MPC canister is constructed from a menu of stainless steels termed Alloy X. These stainless steel materials do not undergo a ductile-to-brittle transition in the minimum service temperature range of the HI-STORM UMAX system. Therefore, brittle fracture is not a concern for the MPC components. Such an assertion cannot be made *a' priori* for the HI-STORM storage overpack and HI-TRAC transfer cask that contain ferritic steel parts. In general, the impact testing requirement for the HI-STORM overpack and the HI-TRAC transfer cask is a function of two parameters: the Lowest Service Temperature (LST)† and the normal stress level. The significance of these two parameters, as they relate to impact testing of the VVM is discussed below.

In normal storage mode, the LST of the HI-STORM storage overpack structural members may reach -40°F in the limiting condition wherein the spent nuclear fuel (SNF) in the contained MPCs emits no (or negligible) heat and the ambient temperature is at -40°F (design minimum per Chapter 2: Principal Design Criteria). During the heavy load handling operations at an ISFSI, the applicable lowest service temperature in the MPC's in FW docket is limited to a threshold ambient temperature below which lifting and handling of the HI-TRAC transfer cask is not permitted by the Technical Specification. Therefore, two distinct LSTs are applicable to load bearing metal parts within the HI-STORM UMAX overpack and the HI-TRAC transfer cask; namely,

LST = 0°F for the HI-STORM overpack during handling operations and for the HI-TRAC transfer cask during all normal operating conditions.

LST = -40°F for the HI-STORM overpack during all non-handling operations (i.e., normal and off-normal storage mode).

Parts used to lift the overpack or the transfer cask, which include the top flange in HI-TRAC, are referred to as "significant-to-handling" (STH) parts. All other parts of the overpack and the transfer cask will be referred to as "NF" components. It is important to ensure that all materials designated as "NF" or "STH" parts possess sufficient fracture toughness to preclude brittle fracture. For the STH parts, the necessary level of protection against brittle fracture is deemed to exist if the NDT (nil ductility transition) temperature of the part is at least 40°F below the LST.

It is well known that the NDT temperature of steel is a strong function of its composition, manufacturing process (viz., fine grain vs. coarse grain practice), thickness, and heat treatment.

* This subsection has been copied from the HI-STORM 100 FSAR (Section 3.1) without any substantive change.
† LST (Lowest Service Temperature) is defined as the daily average for the host ISFSI site when the outdoors portions of the "short-term operations" are carried out.

For example, it is well known that increasing the carbon content in carbon steels from 0.1% to 0.8% leads to the change in NDT from -50°F to approximately 120°F. Likewise, lowering of the normalizing temperature in the ferritic steels from 1200°C to 900°C may lower the NDT from 10°C to -50°C. It therefore follows that the fracture toughness of steels can be varied significantly within the confines of the ASME Code material specification set forth in Section II of the Code. For example, SA516 Gr. 70 can have a maximum carbon content of up to 0.3% in plates up to four inches thick. Section II further permits normalizing or quenching followed by tempering to enhance fracture toughness. Manufacturing processes that have a profound effect on fracture toughness, but little effect on tensile or yield strength of the material, are also not specified with the degree of specificity in the ASME Code to guarantee a well-defined fracture toughness. In fact, the Code relies on actual coupon testing of the part to ensure the desired level of protection against brittle fracture. For Section III, Subsection NF Class 3 parts, the desired level of protection is considered to exist if the lowest service temperature is equal to or greater than the NDT temperature (per NF 2311(b)(10)). Accordingly, the required NDT temperature for all load bearing metal parts in the HI-STORM UMAX overpack (NF and STH) is set at -40°F below the LST.

The STH components (Closure Lid strong backs) have thicknesses less than or equal to 1". The strong backs are fabricated from normalized SA516 Gr.70 material which is exempted from impact testing at lowest service temperatures above -30°F per ASME Section III, Subsection NF. Because the HI-TRAC Transfer Cask operations are limited to a minimum service temperature of 0°F for handling, the lid will also not be handled at temperatures below 0°F and the strong back material is thereby exempt from testing.

All other steel structural materials in the HI-STORM UMAX are made of SA516 Gr. 70, SA515 Gr. 70, SA36 or austenitic stainless steel. The SA516 Gr. 70 material used to fabricate the HI-STORM UMAX is exempt from impact testing per NF-2311(b), because:

i. The LST for handling operations is above the Minimum Design Temperature of SA516 Gr. 70 (for thickness less than 2-1/2") per Figure NF-2311(b)-1, and;

ii. During non-handling operations (i.e., <u>normal</u> storage mode), the maximum tensile stress in the HI-STORM overpack is less than the threshold limit of 6,000 psi specified in NF-2311(b)(7).

If the SA516 Gr. 70 plate is as-rolled (i.e., not normalized), then impact testing is required except when the maximum stress under normal conditions, including handling operations, does not exceed 6,000 psi tension (see NF-2311(b)(7)) or is non-tensile (compressive).

Table 3.1.9 provides a summary of impact testing requirements to ensure prevention of brittle fracture.

8.4.4 Creep Properties of Materials

Creep, a visco-elastic and visco-plastic effect in metals, manifests itself as a monotonically increasing deformation if the metal part is subjected to stress under elevated temperature. Since certain parts of the HI-STORM UMAX system, notably the fuel basket, operate at relatively high temperatures, creep resistance of the fuel basket is an important property. Creep resistance of the MPC internals is discussed in the HI-STORM FW FSAR. Creep is not a concern in the Enclosure Vessel, the HI-STORM UMAX, or the HI-TRAC steel weldment because of the operating metal temperatures, stress levels and material properties. Steels used in ASME Code pressure vessels have a high threshold temperature at which creep becomes a factor in the equipment design. The ASME Code Section II material properties provide the acceptable upper temperature limit for metals and alloys acceptable for pressure vessel service. In the selection of steels for the HI-STORM UMAX system, a critical criterion is to ensure that the sustained (normal) metal temperature of the part made of the particular steel type shall be less than the Code permissible temperature for pressure vessel service. This criterion guarantees that excessive creep deformation will not occur in the steels used in the HI-STORM UMAX system.

8.5 WELDING MATERIAL AND WELDING SPECIFICATION

Welds in the HI-STORM UMAX system are divided into two broad categories:

i. Structural welds
ii. Non-structural welds

Structural welds are those that are essential to withstand mechanical and inertial loads exerted on the component under normal storage and handling.

Non-structural welds are those that are subject to minor stress levels and are not critical to the safety function of the part. Non-structural welds are typically located in the redundant parts of the structure. The guidance in the ASME Code Section NF-1215 for secondary members may be used to determine whether the stress level in a weld qualifies it to be categorized as non-structural.

Both structural and non-structural welds must satisfy the material considerations listed in Tables 8.1.1 and 8.1.2 for the MPC and the HI-STORM UMAX VVM, respectively. In addition, the welds must not be susceptible to any of the applicable failure modes listed in Table 8.1.4
The welding material and welding specification considerations for the MPC and HI-TRAC are discussed in Section 8.5 of the HI-STORM FW FSAR.

To ensure that all structural welds in the HI-STORM UMAX system shall render their intended function, the following requirements are observed:

i. The welding procedure specifications comply with ASME Section IX for every Code material used in the system.

ii. The quality assurance requirements applied to the welding process correspond to the highest ITS classification of the parts being joined.

iii. The non-destructive examination of every weld is carried out using quality procedures that comply with ASME Section V.

The welding operations are performed in accordance with the requirements of codes and standards depending on the design and functional requirements of the components.

The selection of the weld wire, welding process, range of essential and non-essential variables,* and the configuration of the weld geometry has been carried out to ensure that each weld will have:

i. Greater mechanical strength than the parent metal.

* Please refer to Section IX of the ASME Code for the definition and delineation of essential and non-essential variables.

ii. Acceptable ductility, toughness, and fracture resistance.
iii. Corrosion resistance properties comparable to the parent metal.
iv. No risk of crack propagation under the applicable stress levels.

The welding procedures implemented in the manufacturing of HI-STORM UMAX components are intended to fulfill the above performance expectations.

The inspection and testing requirements of the HI-STORM UMAX System component welds are provided in Section 10.1.

The weld filler material shall comply with requirements set forth in the applicable Welding Procedure Specifications qualified to ASME Section IX at the manufacturer's facility. Only those Welding Procedures that have been qualified to the Code are permitted in the manufacturing of HI-STORM UMAX components.

The weld procedure qualification record specifies the requirements for fracture control (e.g., post weld heat treatment). The HI-STORM UMAX module assembly does not require any post weld heat treatment due to the material combinations and provisions in the applicable codes and standards.

Non-structural welds shall meet the following requirements:

1. The welding procedure shall comply with Section IX of the ASME Code or AWS D1.1.

2. The welder shall be qualified, at minimum, to the commercial code such as ASME Section VIII, Div.1, or AWS D1.1.

3. The weld shall be visually examined by the weld operator or a Q.C. inspector qualified to Level 1 (or above) per ASNT designation.

8.6 BOLTS AND FASTENERS

The HI-STORM UMAX VVM assembly does not employ any ITS bolts or fasteners. However, during the MPC transfer into the HI-STORM UMAX, the HI-TRAC and mating device may be attached to the VVM assembly to prevent tip-over during a seismic event. If bolts are used to secure the HI-TRAC against tip-over, the bolts and anchor location material would be considered ITS and would be procured in accordance with the Holtec QA program. Bolt and anchor location material would be selected from the list of materials identified in ASME Section II. The bolting materials used for the HI-TRAC are evaluated in section 8.6 of the HI-STORM FW FSAR.

The only bolts employed in the HI-STORM UMAX VVM system are those used to secure the vent flue to the inlet and outlet plenums. All bolts and fasteners are made of alloy materials which are not expected to experience any significant corrosion in the operating environment. The ISFSI operation and maintenance program shall call for coating of bolts and fasteners if the ambient environment is aggressive.

All threaded surfaces are treated with a preservative to prevent corrosion. The O&M program for the storage system calls for all bolts to be monitored for corrosion damage and replaced, as necessary.

8.7 COATINGS AND CORROSION MITIGATION

In order to provide reasonable assurance that the VVM will meet its intended Design Life of 60 years and perform its intended safety function(s), chemical and galvanic reactions and other potentially degrading mechanisms must be accounted for in its design and construction. Coatings and corrosion mitigation techniques related to the MPC and HI-TRAC are discussed in Section 8.7 of the HI-STORM FW FSAR.

It should be noted that, although the CEC is a buried steel structure it is substantially sequestered from the native soil through two engineered features:

a. A thick reinforced concrete Enclosure Wall surrounds the VVM array and, along with the Support Foundation pad, provides a physical separation (water intrusion protection) to the CECs.

b. The subgrade in contact with the CECs is either a "free flow" concrete or an engineered fill selected to provide a non-aggressive environment around the CECs.

The above engineered features provide an environmentally benign condition for the CECs. The above said, although the CEC is not a part of the MPC confinement boundary, it should not corrode to the extent where localized in-leakage of water occurs or where gross general corrosion prevents the component from performing its primary safety function. In the following, considerations in the VVM's design and construction consistent with the applicable guidance provided in ISG-15 [8.1.1] are summarized.

All VVM components are protected from galvanic corrosion by appropriate designs. Except for the CEC exterior surfaces (exterior CEC surface coating requirements discussed separately), all carbon steel surfaces of the VVM are lined and coated with the same or equivalent surface preservative that is used in the aboveground HI-STORM FW and HI-STORM 100 overpacks. The pre-approved surface preservative is a proven zinc-rich inorganic/metallic (may also be an organic zinc rich coating) material that protects galvanically and has self-healing characteristics for added protection. All exposed surfaces interior to the VVM are accessible for the reapplication of surface preservative, if necessary.

Additional preemptive measures to prevent corrosion are essential, if the substrate is of aggressive chemistry. A description of corrosion mitigation measures proposed to protect the HI-STORM UMAX systems and which are also approved for use in HI-STORM 100U VVM in Docket Number 72-1014, are presented in the following.

The native soil excavated at the ISFSI site shall preferably not be used as subgrade unless it has the requisite density and low corrosivity. To evaluate soil corrosivity, a "10 point" soil-test evaluation procedure, in accordance with the guidelines of Appendix A of ANSI/AWWA C105/A21 [8.7.1] will be utilized. The classical soil evaluation criteria in the aforementioned standard focuses on parameters such as: (1) resistivity, (2) pH, (3) redox (oxidation-reduction) potential, (4) sulfides, (5) moisture content, (6) potential for stray current, and (7) experience

with existing installations in the area. Using the procedure outlined in ref.[8.7.1] , the ISFSI soil environment corrosivity is categorized as either "mild" for a soil test evaluation resulting in 9 points or less or "aggressive" for a soil test evaluation resulting in 10 points or greater. The following table details the corrosion mitigation measures that shall be necessary if the native soil is used as the subgrade:

Implementation of Corrosion Mitigation Measures

Soil Environment Corrosivity	**Corrosion Mitigation Measures**		
	Coating (see note (i))	Concrete Encasement (see note (ii))	Cathodic Protection (see note (iii))
Mild	Required	Choice of either concrete encasement or cathodic protection; or both	
Aggressive	Required	Optional	Required

Notes:
i. An acceptable exterior surface preservative (coating) applied on the CEC carbon steel surface.
ii. Concrete encasement of the CEC external surfaces to establish a high pH buffer around the metal mass.
iii. A suitably engineered impressed current cathodic protection system (ICCPS).

The corrosion mitigation measures tabulated above are further detailed in the following subsections:

i. Exterior Coating

The CEC exterior shall be coated with a radiation resistant surface preservative designed for below-grade and/or immersion service. Inorganic and/or metallic coatings are sufficiently radiation-resistant for this application; therefore, radiation testing is not required. Organic coatings such as epoxy, however, must have proven radiation resistance or must be tested without failure to at least 10^7 Rad. Radiation resistance to lower radiation levels is acceptable on a site-specific basis. Radiation testing shall be performed in accordance with ASTM D 4082 [8.7.6] or equivalent. The coating should be conservatively treated as a Service Level II coating as described in Reg. Guide 1.54 [8.7.3]. As such, the coating shall be subjected to appropriate quality assurance in accordance with the applicable guidance provided by ASTM D 3843-00 [8.7.4]. The coating should preferably be shop-applied in accordance with manufacturer's instructions and, if appropriate, applicable guidance from ANSI C 210-03 [8.7.5]. The following table provides the acceptance criteria for the selection of coatings for the exterior surfaces of the CEC and ranks them in order of importance.

Acceptance Criteria for the Selection of Coatings	
Rank	**Criteria**
1	suitable for immersion and/or below grade service
2a	compatible with the ICCPS (if used) • adequate dielectric strength • adequate resistance to cathodic disbondment
2b	compatible with concrete encasement (if used) • adequate resistance to high alkalinity
3	adequate radiation resistance
4	adequate adhesion to steel
5	adequate bendability/ductility/cracking resistance/abrasion resistance
6	adequate strength to resist handling abuse and substrate stress

The Keeler & Long polyamide-epoxy coating is selected as one that satisfies the acceptance criteria to the maximum degree. Alternatively, a Holtec-approved equivalent that meets the acceptance criteria set forth in the table above may be used.

ii. Concrete Encasement

The CEC concrete encasement shall provide a minimum of 5 inches of cover to provide a pH buffering effect for additional corrosion mitigation. The required 5-inch minimum thickness is more conservative than that recommended in ACI Codes, such as ACI 318, which call for up to 3 inches of concrete cover over steel reinforcement in aggressive environments. Considering that the concrete encasement is restricted to mild soil environments (unless used in conjunction with cathodic protection) and has a non-structural role, an approximately 5-inch thick concrete encasement is considered more than sufficient to provide reasonable assurance that the design basis service life can be achieved. The lowest part of the CEC sits on the Support Foundation Pad.

Regardless of reinforcement method, the material selected shall be corrosion-resistant or otherwise appropriately coated (e.g., epoxy coated steel wire) for corrosion resistance.

The concrete encasement shall be installed in accordance with Holtec-approved procedures that utilize applicable guidance from the ACI codes (e.g., ACI 318 for commercial concrete). Installation procedures shall address mix designs (incorporating Portland cement), testing, mixing, placement, and reinforcement, with the aim to enhance concrete durability and minimize voids and micro-cracks.

iii. Impressed Current Cathodic Protection System (ICCPS)

If the aggressiveness of the subgrade around the CEC is highly aggressive and warrants an ICCPS then the user may choose to either extend an existing ICCPS to protect the installed ISFSI, or to establish an autonomous system. The initial startup of the ICCPS must occur within one year after installation of the VVM to ensure timely corrosion mitigation. In addition, the ICCPS should be maintained operable at all times after initial startup except for system shutdowns due to power outages, repair or preventive maintenance and testing, or system modifications. Because there are a multitude of ISFSI variables that will bear upon the design of the ICCPS for a particular site, the essential criteria for its performance and operational characteristics are set down in this FSAR, which the detailed design work for each ISFSI site must follow.

Design Criteria for the Impressed Current Cathodic Protection System
a. The cathodic protection system shall be capable of maintaining the CEC at a minimum (cathodic) potential as required by NACE Standard RP0285-2002 [8.7.7].   b. The ICCPS shall include provisions to infer its proper operation and effectiveness on a periodic basis.   c. The system shall be designed to mitigate corrosion of the CEC for its design life. Alternatively, the system shall be designed to mitigate corrosion of the CEC for its License Life with due consideration to provisions that would allow system upgrades or maintenance for extended service as required for potential future license extensions.   d. The cathodic protection system design, installation, operation, testing, and maintenance shall follow the applicable guidelines of: - 49CFR195 Subpart H “Corrosion Control”, Oct. 1, 2004 edition [8.7.8]   - NACE Standard RP0285-2002 “Corrosion Control of Underground Storage Tank Systems by Cathodic Protection” [8.7.7]

The following standards and/or publications may also be utilized for additional guidance in the design, installation, operation, testing, and maintenance of the ICCPS as needed (in case of conflict, the guidelines of item d above shall prevail):

- API RP1632, “Cathodic Protection of Underground Petroleum Storage Tanks and Piping Systems”
- NACE RP0169-96, “Control of External Corrosion on Underground or Submerged Piping Systems”
- 49CFR192 Subpart I “Requirements for Corrosion Control”, Oct. 1, 2004 edition
- Other standards or publications referenced by any of the above three standards and publications.

Records of system operating data necessary to adequately track the operable status of the ICCPS shall be maintained in accordance with the user's quality assurance program.

Finally, the surface preservative used to coat the CEC must meet the requirements described in (i) above but must also be compatible with cathodic protection and resistant to the alkaline conditions created by cathodic protection and/or concrete encasement.

8.8 GAMMA AND NEUTRON SHIELDING MATERIALS

Gamma and neutron shield materials in the HI-STORM UMAX VVM system are discussed in Section 1.2. The gamma and neutron shielding materials used in the MPC and HI-TRAC are discussed in Sections 8.8 and 8.9 of the HI-STORM FW FSAR. The primary shielding materials used in the HI-STORM UMAX VVM system, as listed in Table 8.1.3, are plain concrete, reinforced concrete, and steel.

The plain concrete provides the main shielding function in the HI-STORM UMAX lids to minimize sky shine.

8.8.1 Plain Concrete

Unlike the above ground HI-STORM models, the use of plain concrete for shielding purposes in the underground VVMs is limited to the VVM Closure Lid. The *critical characteristics* of concrete used in the Closure Lid are its density and compressive strength. Table 2.3.2 provides reference properties of plain concrete used in the Closure Lid. The temperature limits in Table 2.3.7 are adopted from the HI-STORM 100 FSAR where the bases for the thermal limits are documented.

The density of plain concrete within the HI-STORM UMAX overpack is subject to a minor decrease due to long-term exposure to elevated temperatures. The reduction in density occurs primarily due to liberation of unbonded water by evaporation.

The shielding analysis in Section 5.3 considers the density of concrete in the Closure Lid as an explicit input item in the computation of dose above the HI-STORM UMAX ISFSI structure.

As stated in sub-section 1.2.2, lean concrete may also be used as the material of construction for the subgrade (space A in Figure 2.4.4). The required structural properties of the subgrade concrete, if used, are provided in Tables 2.3.2 and 3.3.5. Because the subgrade material is completely protected from the ambient environment, the environmental demands on it are not very severe. At sites with high water tables, an Enclosure Wall may be incorporated in the ISFSI design to protect the subgrade from long term erosion and degradation.

8.8.2 Steel

Section 5.3 provides a discussion on steel as a shielding material and its composition used in the evaluation of its shielding characteristics.

8.9 NEUTRON ABSORBING MATERIALS

The discussion related to the selection and use of neutron absorbing materials used in the MPC and HI-TRAC are discussed in Section 8.9 of the HI-STORM FW FSAR and are not repeated here.

8.10 SEALS

The HI-STORM UMAX VVM assembly does not utilize any gaskets that seal against a large pressure differential.

The only external gasket used in the system is the soft gasket at the Closure lid-CEC Flange interface that helps prevent the ingress of moisture and insects (through the small crack that may exist due to weld distortion in the fabrication of interfacing fabricated steel weldment surfaces) into the module cavity space.

The Divider shell is sealed against the Closure lid using a pliable, non-organic seal material that is suitable for long-term ambient air application up to 300 deg. F.

The seals used with the HI-TRAC bottom lid are discussed in Section 8.11 of the HI-STORM FW FSAR.

8.11 CHEMICAL AND GALVANIC REACTIONS

The materials used in the HI-STORM UMAX System are examined to establish that these materials do not participate in any chemical or galvanic reactions when exposed to the various environments during all normal operating conditions and off-normal and accident events. Chemical and galvanic reactions related to the MPC and HI-TRAC are discussed in Section 8.12 of the HI-STORM FW FSAR.

The following acceptance criteria for chemical and galvanic reactions are extracted from ISG-15 [8.1.1] for use in HI-STORM UMAX VVM components.

a. The DCSS should prevent the spread of radioactive material and maintain safety control functions using, as appropriate, noncombustible and heat resistant materials.

b. A review of the DCSS, its components, and operating environments (wet or dry) should confirm that no operation (e.g., short-term loading/unloading or long-term storage) will produce adverse chemical and/or galvanic reactions, which could impact the safe use of the storage cask.

c. Components of the DCSS should not react with one another, or with the cover gas or spent fuel, in a manner that may adversely affect safety. Additionally, corrosion of components inside the containment vessel should be effectively prevented.

d. Potential problems from general corrosion, pitting, stress corrosion cracking, or other types of corrosion, should be evaluated for the environmental conditions and dynamic loading effects that are specific to the component.

The materials and their ITS pedigree are listed in the drawing package provided in Section 1.5. The compatibility of the selected materials with the operating environment and to each other for potential galvanic reactions is discussed in this section.

- External atmosphere – During long-term storage the casks are exposed to outside atmosphere, air with temperature variations, solar radiation, rain, snow, ice, etc.

As discussed herein, the components of the HI-STORM UMAX System have been engineered to ensure that the environmental conditions expected to exist at nuclear power plant installations do not prevent the cask components from rendering their respective intended functions.

The principal operational considerations that bear on the adequacy of the storage overpack for the service life are addressed as follows:

Exposure to Environmental Effects

All exposed surfaces of the HI-STORM UMAX VVM components are made from stainless steels or ferritic steels that are readily painted. Concrete, which serves strictly as a shielding

material in the VVM Closure Lid, is encased in steel. Therefore, the potential of environmental vagaries such as spalling of concrete are ruled out for HI-STORM UMAX VVM. Under normal storage conditions, the bulk temperature of the HI-STORM UMAX storage overpack will change very gradually with time because of its large thermal inertia. Therefore, material degradation from rapid thermal ramping conditions is not credible for the HI-STORM UMAX VVM. Similarly, corrosion of structural steel embedded in the concrete structures due to salinity in the environment at coastal sites is not a concern for HI-STORM UMAX VVM because it does not rely on rebars (indeed, it contains no rebars). The configuration of the storage VVM assures resistance to freeze-thaw degradation. In addition, the storage system is specifically designed for a full range of enveloping design basis natural phenomena that could occur over the service life of the storage system as catalogued in Section 2.2 and evaluated in Chapter 11.

The ISFSI pad, which is exposed to the elements, shall be subject to a surveillance program to monitor its potential degradation, as discussed in Chapter 10.

Material Degradation

The relatively low neutron flux to which the storage overpack is subjected cannot produce measurable degradation of the cask's material properties and impair its intended safety function. Exposed carbon steel components are coated to prevent corrosion. The ambient environment of the ISFSI storage pad mitigates damage due to exposure to corrosive and aggressive chemicals that may be produced at other industrial plants in the surrounding area.

Maintenance and Inspection Provisions

The requirements for periodic inspection and maintenance of the storage overpack throughout its service life are defined in Section 10.4. These requirements include provisions for routine inspection of the storage overpack exterior and periodic visual verification that the ventilation flow paths of the storage overpack are free and clear of debris. ISFSIs located in areas subject to atmospheric conditions that may degrade the storage cask or canister should be evaluated by the licensee on a site-specific basis to determine the frequency for such inspections to assure long-term performance. In addition, the HI-STORM UMAX system is designed for easy retrieval of the MPC from the storage overpack should it become necessary to perform more detailed inspections and repairs on the storage system.

The above findings are consistent with those of the NRC's Waste Confidence Decision Review [8.11.1], which concluded that dry storage systems designed, fabricated, inspected, and operated in accordance with such requirements are adequate for a 60-year Design and a 100-year service life while satisfying the requirements of 10CFR72.

8.12 FUEL CLADDING INTEGRITY

The discussion related to the fuel cladding integrity during short term operations is presented in Section 8.13 of the HI-STORM FW FSAR and are not repeated here.

8.13 EXAMINATION AND TESTING

Examination and testing are integral parts of manufacturing of the HI-STORM UMAX System components. Examination and testing requirements for the MPC and HI-TRAC are described in Section 8.14 of the HI-STORM FW FSAR. A comprehensive discussion on the examinations and testing that are conducted during the manufacturing process for the HI-STORM UMAX VVM is provided in Chapter 10 wherein the applicable codes and standards are also cited along with the acceptance criteria derived from them.

Post-fabrication inspections are discussed in Section 10.4 as part of the HI-STORM UMAX VVM System maintenance program. Inspections are conducted prior to fuel loading or prior to each fuel handling campaign. Other periodic inspections are conducted during storage.

The HI-STORM UMAX VVM is a passive device with no moving parts. Overpack vent screens are inspected on scheduled intervals for damage, holes, etc. The VVM's external surface, including identification markings, is visually examined on a periodic basis in accordance with the ISFSI's surveillance plan. The temperature monitoring system, if used, is inspected per the licensee's QA program and manufacturer's recommendations.

8.14 REGULATORY COMPLIANCE

The preceding sections describe the materials used in important-to-safety SSCs and the suitability of those materials for their intended functions in the HI-STORM UMAX System.

The requirements of 10CFR72.122(a) are met: The material properties of SSCs important to safety conform to quality standards commensurate with their safety functions.

The requirements of 10CFR72.104(a), 106(b), 124, and 128(a)(2) are met: Materials used for shielding are adequately designed and specified to perform their intended function.

The requirements of 10CFR72.122(h)(1) and 236(h) are met: The design of the DCSS and the selection of materials adequately protect the spent fuel cladding against degradation that might otherwise lead to gross rupture of the cladding by ensuring that the cladding temperature remains below the ISG-11 Rev 3 limits..

The requirements of 10CFR72.236(h) and 236(m) are met: The material properties of SSCs important-to-safety will be maintained during normal, off-normal, and accident conditions of operation as well as short-term operations so the spent fuel can be readily retrieved without posing operational safety problems.

The requirements of 10CFR72.236(g) are met: The material properties of SSCs important-to-safety will be maintained during all conditions of operation so the spent fuel can be safely stored for the specified service life and maintenance can be conducted as required.

The requirements of 10CFR72.236(h) are met: The HI-STORM UMAX System employs materials that are not vulnerable to degradation over time or react with one another during long-term storage.

8.15 REFERENCES

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[8.2.5] "Guide Specification for Controlled Low Strength Materials (CLSM)", National Ready Mixed Concrete Association, Silver Spring, MD.

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[8.3.2] ASME Boiler and Pressure Vessel Code, Section III, Appendices, 2010 edition.

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[8.7.2] 49CFR Part 192 Subpart I "Requirements for Corrosion Control, Title 49 of the Code of Federal Regulations, Oct, 1 2004 Edition (or latest), Office of the Federal Register, Washington, D.C.

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[8.7.4] ASTM D 3843-00, "Standard Practice for Quality Assurance for Protective Coatings Applied to Nuclear Facilities", ASTM International, West Conshohocken, PA.

[8.7.5] ANSI C 210-03, "Liquid Epoxy Coating Systems for the Interior and Exterior of Steel Water Pipes".

[8.7.6] ASTM D4082-10, "Standard Test Method for Effects of Gamma Radiation on Coatings for use in Nuclear Power Plants", ASTM International, West Conshohocken, PA.

[8.7.7] NACE Standard RP0285-2002, "Standard Recommended Practice-Corrosion Control of Underground Storage Tank Systems by Cathodic Protection".

[8.7.8] 49CFR Part 195 Subpart H "Corrosion Control", Title 49 of the Code of Federal Regulations, Oct, 1 2004 Edition, Office of the Federal Register, Washington, D.C.

[8.11.1] 10CFR, Waste Confidence Decision Review, USNRC, September 18, 1990.

CHAPTER 9: OPERATING PROCEDURES

9.0 INTRODUCTION

The operations associated with the use of the HI-STORM UMAX System are quite similar to the operations for the HI-STORM 100U system certified in Docket Number 72-1014. Because the HI-STORM UMAX System's scope of certification is limited to loading pre-certified MPCs (Table 1.2.1) using pre-certified transfer cask (Table 1.2.2), the operations culminating in the arrival of the MPC-bearing HI-TRAC VW transfer cask at the ISFSI are governed by the HI-STORM FW System FSAR [9.6.1]. The description of the operations governed by this FSAR pertains to staging the HI-TRAC transfer cask on the VVM cavity and ends with the removal of the empty transfer cask and replacement of the Closure Lid and installation of other appurtenances to place the storage system in a long-term storage configuration. The necessary information for executing the reverse set of steps to retrieve an MPC from a storage cavity is likewise provided. The guidance provided in this chapter shall be used as an aid to develop the short-term operations procedure specific to a host site that elects to deploy the HI-STORM UMAX system. The procedures provided in this chapter are prescriptive to the extent that they provide the basis and general guidance for plant personnel in preparing detailed written, site-specific, loading, handling, storage, and unloading procedures. Users may add, modify the sequence of, perform in parallel, or delete steps as necessary provided that the intent of this guidance is met and the requirements of the Certificate of Compliance (CoC) are complied within a *literal manner*. The information provided in this chapter complies with the provisions of NUREG-1536.

The information presented in this chapter along with the technical basis of the system design described in this SAR will be used to develop detailed operating procedures. In preparing the site-specific procedures, the user must consult the conditions of the CoC, equipment-specific operating instructions, and the plant's working procedures as well as the information in this chapter to ensure that the short-term operations shall be carried out with utmost safety and ALARA.

The following generic criteria shall be used to determine whether the site-specific operating procedures developed pursuant to the guidance in this chapter are acceptable for use:

- All heavy load handling instructions are in keeping with the guidance in industry standards and Holtec-provided instructions.
- The procedures are in conformance with this FSAR and its CoC.
- The procedures are in conformance with the HI-STORM FW System FSAR [9.6.1].
- The operational steps are ALARA.
- The procedures contain provisions for documenting successful execution of all safety significant steps for archival reference.

- Procedures contain provisions for classroom and hands-on training and for a Holtec-approved personnel qualification process to ensure that all operations personnel are adequately trained.
- The procedures are sufficiently detailed and articulated to enable craft labor to execute them in *literal compliance* with their content.

ISFSI owners are required to develop or modify existing programs and procedures to account for the implementation of the HI-STORM UMAX system. Written procedures are required to be developed or modified to account for such items as handling and storage of systems, structures and components (SSCs) identified as *important-to-safety*, heavy load handling, specialized instrument calibration, special nuclear material accountability, fuel handling procedures, training, equipment, and process qualifications. Users shall implement controls to ensure that all critical set points (e.g., Lift Weights) do not exceed the design limit of the specific equipment.

Control of the operation shall be performed in accordance with the user's Quality Assurance (QA) program to ensure critical steps are not overlooked and that the cask has been confirmed to meet all requirements of the CoC before being released for on-site storage under 10 CFR Part 72.

ALARA warnings highlighted in this chapter are included to alert users to radiological issues. Actions identified with these items are of an advisory nature and shall be implemented based on site-specific determination by the plant's radiation protection personnel.

Section 9.1 provides the technical basis for loading and unloading procedures. Section 9.2 provides the guidance for loading the HI-STORM UMAX system. Section 9.3 provides the procedures for ISFSI operations and general guidance for performing maintenance and for responding to abnormal events that may occur during normal loading operations. Section 9.4 provides the procedure for unloading the HI-STORM UMAX System. The loading steps and the illustrations, are illustrative (rather than definitive) because the architecture of a particular plant and its ISFSI may require significant adaptations for a safe and ALARA loading program.

Because the MPCs and transfer casks utilized in the HI-STORM UMAX system are originally certified in the HI-STORM FW docket, there is a close informational nexus between the two dockets with respect to the components common to both systems. Table 9.0.1 provides a matrix of the information in this FSAR that is supplemented by the corresponding material from the HI-STORM FSAR.the latest QA validated version of which (Rev 1) is placed in this docket for reference purposes.

TABLE 9.0.1: APPLICABLE SECTIONS OF HI-STORM FW FSAR*

Location of UMAX FSAR	**Subject of the reference**	**Location in HI-STORM FW FSAR, Revision 3**
Section 9.0	The operations culminating in the arrival of the MPC-bearing HI-TRAC at the ISFSI	Sections 9.1 and 9.2
Sub-Section 9.2.3	Loading of MPC	Sub-Sections 9.2.1 and 9.2.3
Sub-Section 9.4.1	Direction on further action such as return MPC to the fuel pool or loading in a transport cask for off-site shipment	Section 9.4
Sub-Section 9.4.2	HI-TRAC receipt inspection and cleanliness inspection	Table 9.2.5
Section 9.5	Steps to remove an MPC from a loaded VVM	Sub-Sections 9.4.3 and 9.4.4

* For convenience of reference, the specific revision of the HI-STORM FW FSAR that is referenced in the safety analysis herein is placed in this docket. Updated versions of the HI-STORM FW FSAR shall be placed in this docket as necessary so as to ensure that the safety analyses on the "UMAX" docket (72-1040) remain aligned with the material referenced in the HI-STORM FW FSAR.

9.1 TECHNICAL AND SAFETY BASIS FOR LOADING AND UNLOADING PROCEDURES

The procedures herein are developed for the loading, storing, and unloading of a loaded MPC in the HI-STORM UMAX System. The design of the HI-STORM UMAX System, along with the implementation procedures, the ancillary equipment, and the Technical Specifications, collectively serve to achieve ALARA, minimize risks to the operational staff, and mitigate consequences of potential adverse events.

The primary objective of the information presented in this chapter is to identify and describe the sequence of significant operations and actions that are important-to-safety for canister loading, canister handling, storage operations, and canister unloading to adequately protect crew health and to eliminate any conceivable danger to life or property, to protect the MPC's contents from dispersal, and to provide for the safe execution of tasks and operations.

In the event of an extreme environmental condition, the appropriate procedural guidance to respond to the situation must be available and ready for implementation at the nuclear plant. As a minimum, the procedures shall address establishing emergency action levels, implementation of emergency action program, establishment of personnel exclusions zones, monitoring of radiological conditions, actions to mitigate or prevent the release of radioactive materials, recovery planning and execution, and reporting to the appropriate regulatory agencies, as required.

9.2 PROCEDURE FOR PLACING THE LOADED MPC IN THE HI-STORM UMAX VVM

9.2.1 Overview of Loading Operations

The HI-STORM UMAX System differs from the HI-STORM FW System in that the vertical ventilated module (VVM) is an integral part of the ISFSI and is therefore immovable (cannot be transported on-site). Therefore, the transfer of the MPC from the transfer cask to the storage module cannot occur within the plant's Part 50 structure. Like the HI-STORM 100U the underground module (in Docket Number 72-1014), the transfer of the MPC occurs by staging the HI-TRAC transfer cask over the CEC cavity. The loaded HI-TRAC transfer cask is transported between the ISFSI and the Part 50 facility where the MPC is loaded using a heavy haul vertical cask transporter, heavy haul transfer trailer, a rail car, air pads and the like that have been previously used for such applications. In what follows, the acronym VCT (for vertical cask transporter) is used to denote the hauling machinery. The operational steps required to prepare, load the MPC, and transport it to the ISFSI using the HI-TRAC transfer cask are governed by the HI-STORM FW System FSAR [9.6.1]. The detailed operational steps presented in this chapter, therefore, start with the preparation to stage the loaded HI-TRAC over the recipient storage cavity.

Prior to the start of MPC transfer at the ISFSI, the VVM lid is removed and the storage cavity is inspected for absence of foreign matter. The Mating Device is positioned on top of the VVM. The outlet flue installed on other VVMs on the path to the recipient CEC cavity may need removal for short term operational reasons, such as to enable the VCT to reach the target cavity. It is recommended that an outlet flue be re-installed within approximately 6 hours to minimize thermal impacts to the system. If used, the Supplemental Cooling System (SCS) is disconnected from the HI-TRAC, the HI-TRAC annulus is drained, and the HI-TRAC is placed on top of the Mating Device (Figure 9.2.1). The MPC may be downloaded using the vertical cask crawler, the MPC downloader attached to the transfer cask, or other suitable lifting device. The MPC lifting device is attached to the MPC. The bottom lid is removed and the Mating Device drawer is opened. Optional temporary shielding may be installed, as guided by the licensee's Radiation Protection program, on or around the Mating Device. The MPC is lowered into the VVM (Figure 9.2.2). Following verification that the MPC is fully lowered, the MPC slings are disconnected from the lifting device and lowered onto or removed from the MPC lid. Alternately, the MPC lifting device may be removed. Any temporary shielding is removed, if necessary. The HI-TRAC and bottom lid may be reattached while on the VVM or may be removed and then reattached. HI-TRAC is removed from on top of the VVM. The MPC lift device is removed as necessary. The Mating Device is removed and the VVM lid is installed. Finally, the temperature monitoring elements and their instrument connections, if used at the ISFSI, are installed, and post-loading performance verification specified in the site loading procedures is performed.

Because the HI-STORM UMAX VVM is an integral part of the ISFSI itself, there is no handling

of the VVM. For an example of the rigging required to handle the HI-STORM UMAX VVM Closure Lid, see Figure 9.2.3.

9.2.2 Preparation for MPC Transfer

The required equipment/devices that participate in the transferring of the MPC into dry storage are, as a minimum:

1. Equipment to remove and install the VVM Closure Lid;
2. The vertical cask transporter (VCT) or equivalent load handling devices with redundant drop protection features;
3. The loaded transfer cask containing the MPC;
4. The Mating Device; and
5. MPC lifting and handling devices.

Prior to staging the Mating Device and the transfer cask on the recipient VVM cavity, the storage cavity shall be inspected for absence of debris, water, animals or insect nests, and the like. A general checklist for performing the pre-staging inspection of the VVM cavities is provided below:

1. The painted surfaces shall be inspected for corrosion and chipped, cracked, or blistered paint.
2. All lid surfaces shall be relatively free of dents, scratches, gouges, or other damage.
3. Lid lifting points shall be inspected for dirt, debris, and general condition.
4. Vent openings shall be free from obstructions.
5. Vent screens shall be available, intact, and free of holes and tears.
6. Temperature monitoring elements, if used, shall be inspected for availability, function, calibration, and provisions for mounting to the VVM outlet air passage.

<u>HI-STORM UMAX VVM Main Body</u>

1. Cooling passages shall be free from obstructions.
2. The interior cavity shall be free of debris, litter, tools, and equipment.
3. Painted surfaces shall be inspected for corrosion, and chipped, cracked or blistered paint.

<u>VERTICAL CASK TRANSPORTER (VCT)</u>

The VCT shall be serviced before the beginning of a dry storage campaign and all VCT checks are performed in accordance with its manufacturer's O&M manual. The quantity of fuel and other combustibles in the VCT shall be confirmed to be within the limits specified in the site's 72.212 safety evaluation report. The VCT shall be operated only if the ambient temperature is within the specified limit in the VCT's O&M manual. The VCT operator must have received training in the use of the VCT as specified in its O&M manual.

Finally, the Mating Device shall be inspected for any loose paint, scale, or other adherent debris and cleaned appropriately to eliminate the risk of foreign materials adhering to it that fall into the VVM cavity. The smooth operation of the lid drawer traction system shall be confirmed and moving surfaces lubricated, as specified in the applicable Holtec International Purchasing Specification.

The MPC transfer shall not occur if the meteorological forecast indicates a credible chance of adverse weather activity such as lightning, snow fall, rain, or heavy winds at the ISFSI during the planned transfer operation.

All slings and fasteners shall be inspected for general condition and indication of wear and degradation (such as a cut or nick in the sling) before accepting them for use.

9.2.3 Placement of the MPC into Storage

The operational steps presented in the following are intended to provide guidance to the user in preparing site-specific loading procedures that must be prepared with due consideration of the particular site's physical characteristics, its rigging plan, and its safety/ALARA practices. Figure 9.2.4 provides a pictorial overview of the loading steps.

Loading Steps

1. Load the MPC in accordance with Chapter 9 of the HI-STORM FW System FSAR [9.6.1]
2. Remove the Closure Lid using a crane or other equivalent lifting device.
3. Install the Mating Device on the VVM.
4. Place the loaded HI-TRAC transfer cask recipient on the Mating Device using the vertical cask crawler (or other suitable transportation device).
5. Rig the MPC to the lift components of the cask transporter or other suitable downloading device. Raise the MPC slightly to remove the weight of the MPC from the HI-TRAC Bottom lid (also called the Pool lid).
6. Unbolt the Bottom lid from the HI-TRAC body and lower the lid into the Mating Device.
7. Open the Mating Device drawer.

ALARA Warning:
Temporary shielding may be used to reduce personnel dose during transfer operations. If ALARA considerations dictate that temporary shielding not be used, personnel must remain clear of the immediate area around the Mating Device drawer during MPC downloading.

8. At the user's discretion, install temporary shielding to cover the potential streaming paths around the Mating Device drawer.
9. Lower the MPC into the VVM.
10. Verify that the MPC is fully seated in the VVM.

Caution:
Operations steps that occur with the MPC in the VVM with the Mating Device installed and the drawer closed must be performed in an expeditious manner to avoid excessive heating of the MPC and fuel. The Mating Device must be removed or the drawer opened to establish air cooling within the time limits described in Section 4.5. In the event of equipment malfunction that results in the blockage of air flow, corrective actions must occur within the time limits of the 100% blocked duct accident condition.

11. Disconnect the MPC rigging from the downloading device and lower them onto the MPC lid or remove them from the MPC.
12. Remove any temporary shielding and close the Mating Device drawer.
13. Reinstall the HI-TRAC Bottom lid.

ALARA Warning:
Personnel should remain clear (to the maximum extent practicable) of the VVM annulus when HI-TRAC is being removed to comply with ALARA requirements.

14. Remove the HI-TRAC transfer cask from the top of the VVM.
15. Partially open the Mating Device drawer and remove any MPC rigging.
16. DELETED
17. Close the Mating Device drawer and remove the Mating Device from on top of the VVM.

Guidance:
The lid shall be preferably kept less than 2 feet above the top surface of the VVM while over the MPC. This lift limit action is purely a defense-in-depth measure because the Closure Lid cannot fall and impact the MPC because of geometric constraints.

18. Install the VVM lid. Check that the rigging (in its specific configuration) is rated to lift the load (rated to lift two times the load per NUREG 0612).
19. Remove the VVM lid rigging equipment and re-install the outlet vent cover (if previously removed).
20. Install the VVM temperature monitoring elements (if used).
21. Install the flue extensions (that were removed to avoid interference with the VCT).
22. Perform shielding effectiveness testing, if required by the Technical Specification..

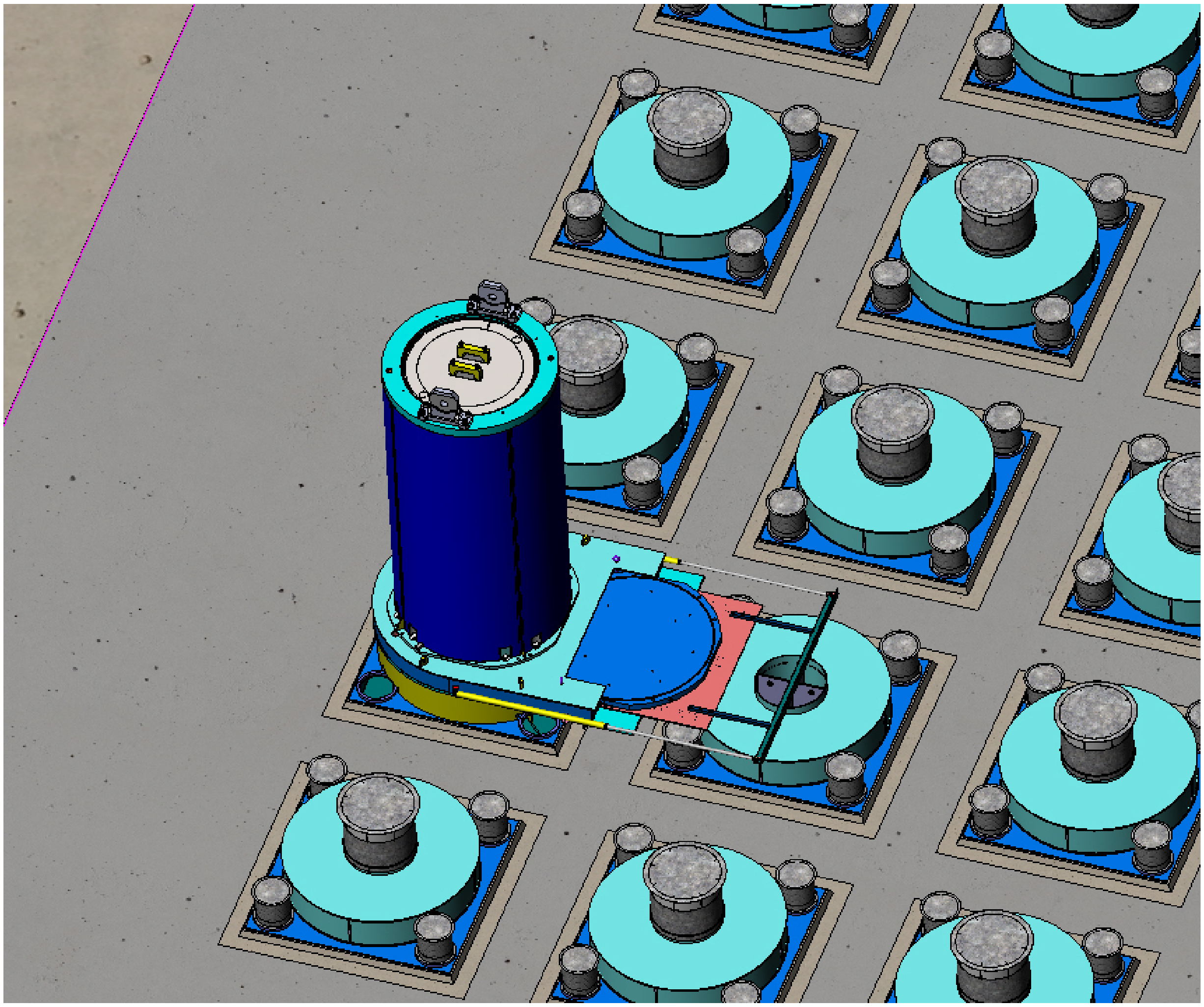


FIGURE 9.2.1: HI-TRAC ALIGNMENT AND PLACEMENT ON MATING DEVICE AND HI-STORM UMAX VVM

Note: The design features of the HI-STORM UMAX System are the exclusive intellectual property of Holtec International under U.S. and international patent right laws. Minor details of the HI-STORM UMAX depicted here and other figures in this FSAR may vary slightly from the licensing drawings in Section 1.5.

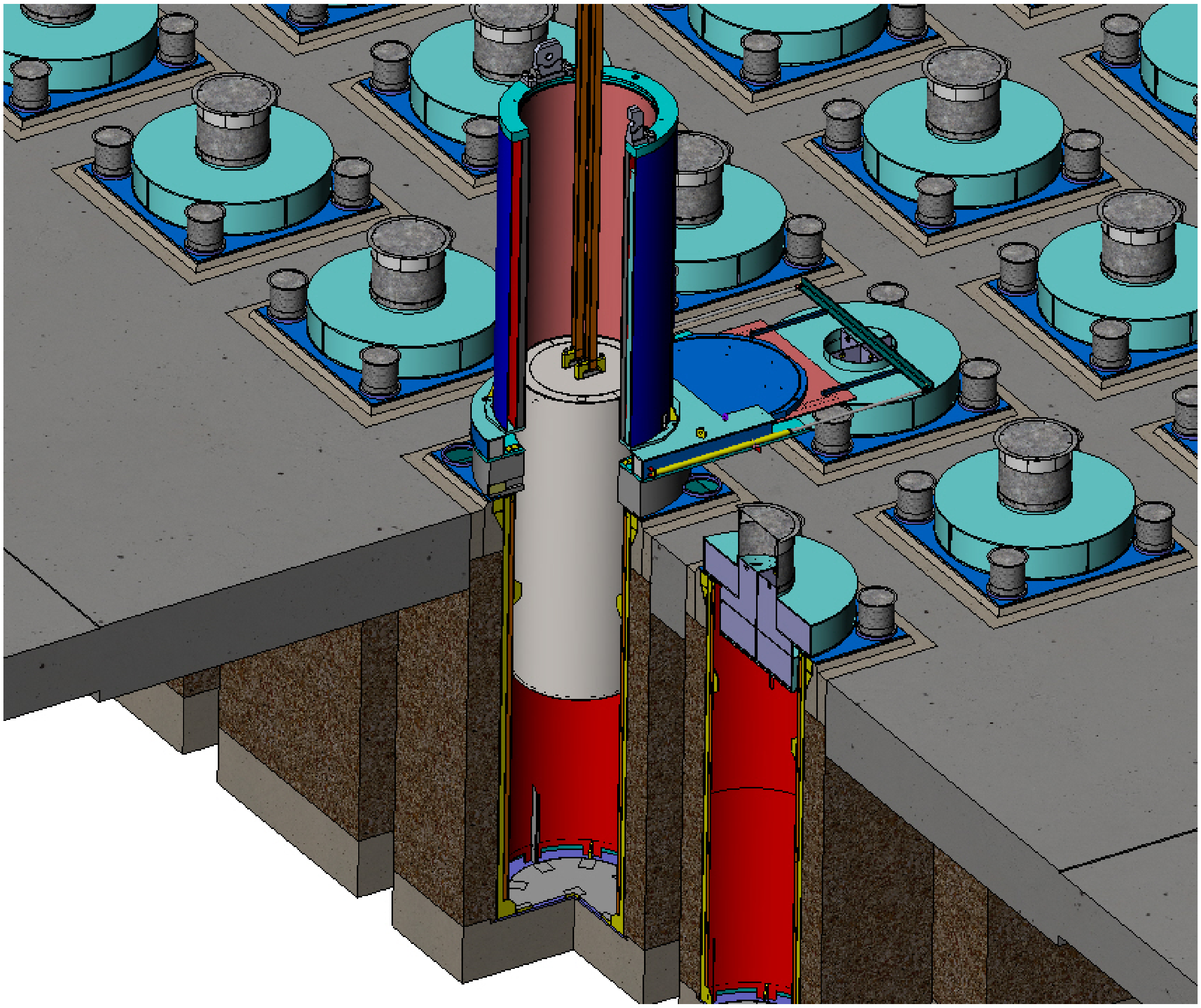


FIGURE 9.2.2: DOWNLOADING MPC INTO HI-STORM UMAX VVM

Note: The design features of the HI-STORM UMAX System are the exclusive intellectual property of Holtec International under U.S. and international patent right laws. Minor details of the HI-STORM UMAX depicted here and other figures in this FSAR may vary slightly from the licensing drawings in Section 1.5.

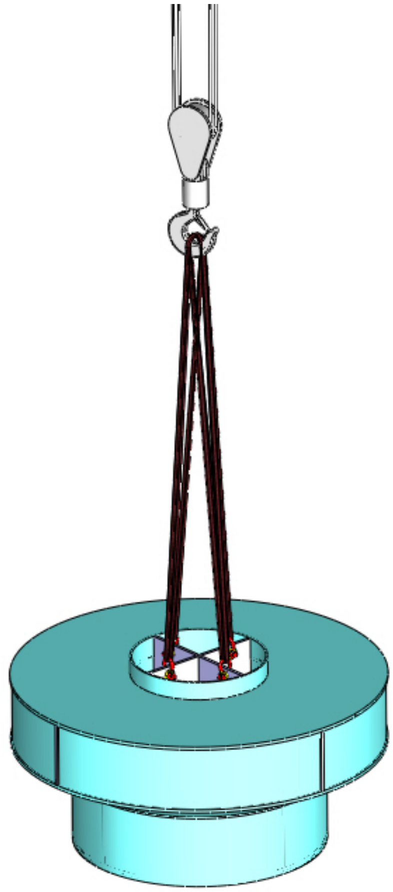
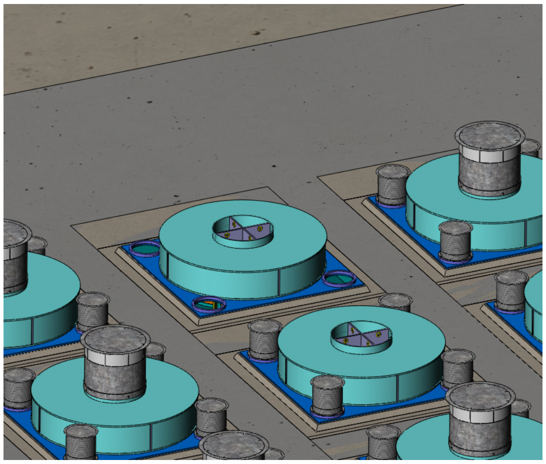


FIGURE 9.2.3: ILLUSTRATIVE RIGGING CONFIGURATION FOR THE HI-STORM UMAX VVM LID

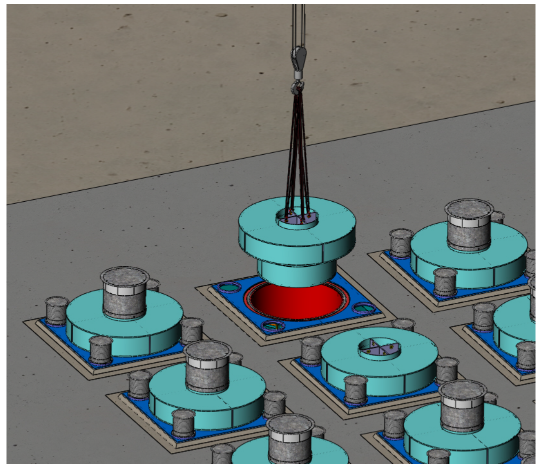
Note: The design features of the HI-STORM UMAX System are the exclusive intellectual property of Holtec International under U.S. and international patent right laws. Minor details of the HI-STORM UMAX depicted here and other figures in this FSAR may vary slightly from the licensing drawings in Section 1.5.

A.



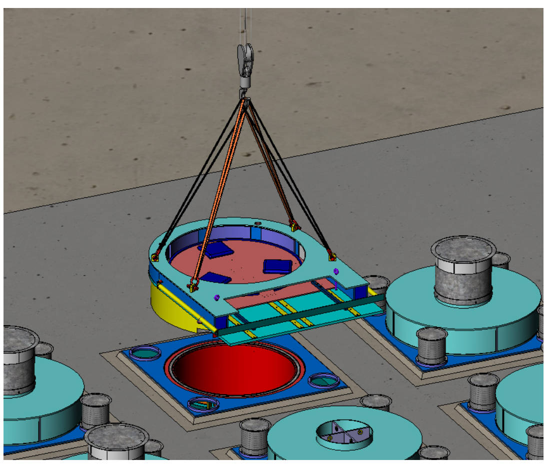
Remove the flue extensions to access the lid lifting features.

B.



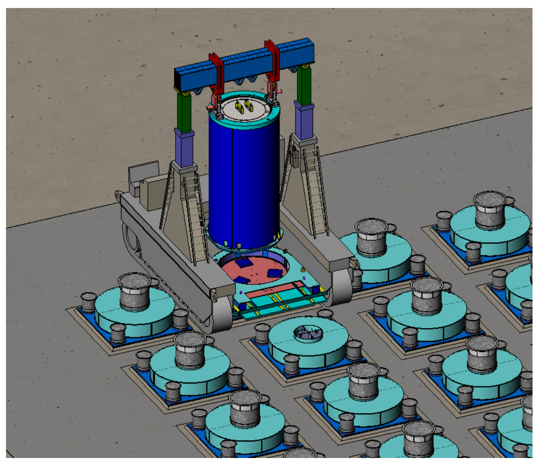
Remove the UMAX lid from the VVM to be loaded.

C.



Install the mating device on the VVM to be loaded.

D.



Align the HI-TRAC to the mating device using the vertical cask transporter.

FIGURE 9.2.4: PICTORIAL OVERVIEW OF THE LOADING STEPS
(SHEET 1 OF 3)

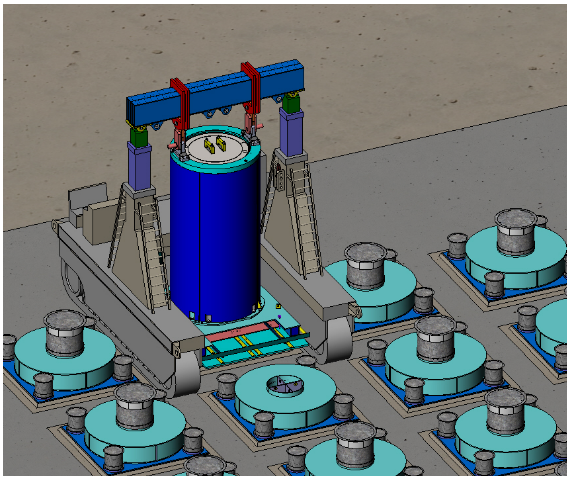
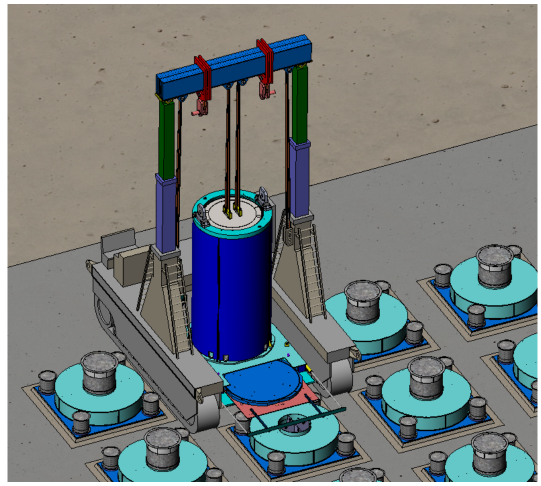
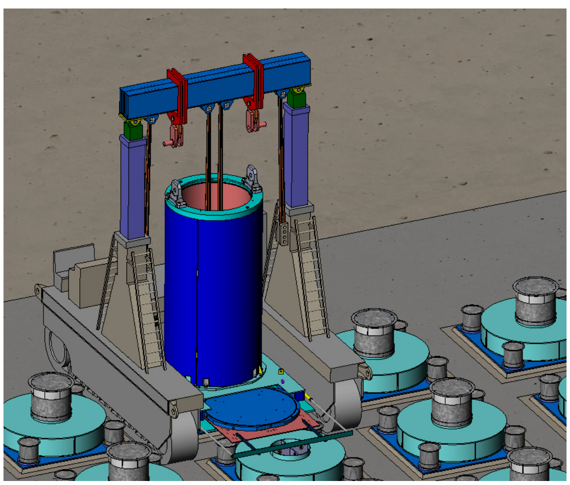
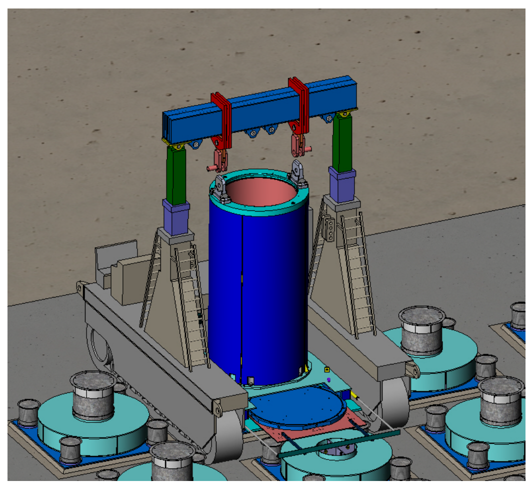
E.  **Mate the HI-TRAC to the VVM.**	**F.**  **Attach the MPC rigging to the vertical cask transporter.** **Raise the MPC slightly.** **Remove the HI-TRAC bottom lid bolts.** **Open the mating device.**
G.  **Lower the MPC into the VVM.**	**H.**  **Disconnect the rigging from the vertical cask transporter and lower it onto the MPC lid.**

FIGURE 9.2.4: PICTORIAL OVERVIEW OF THE LOADING STEPS
(SHEET 2 OF 3)

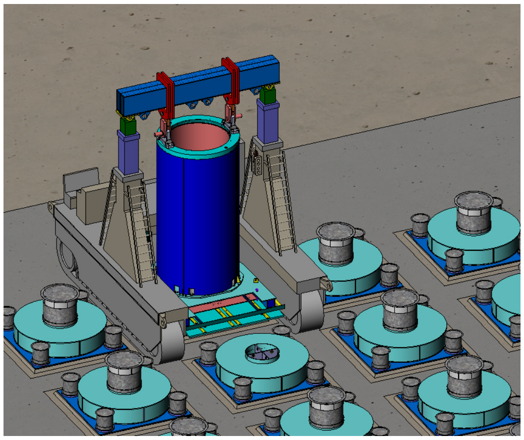
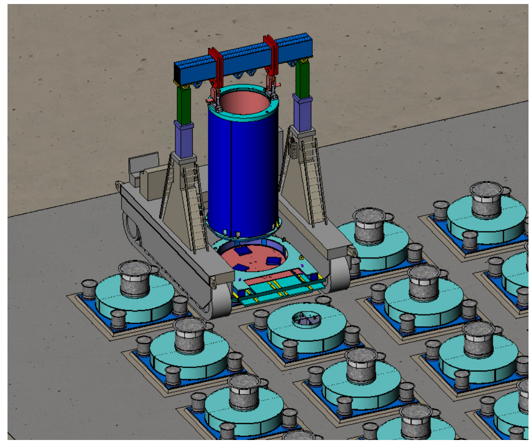
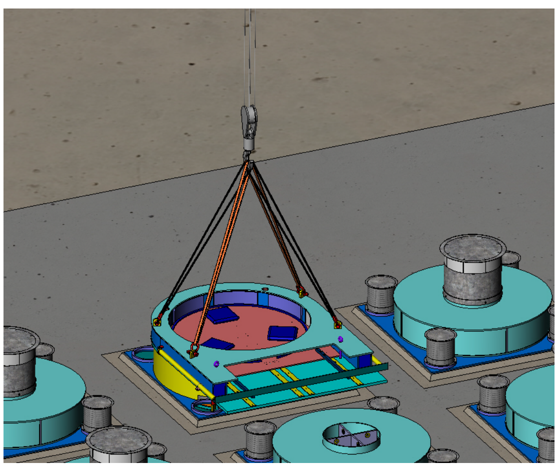
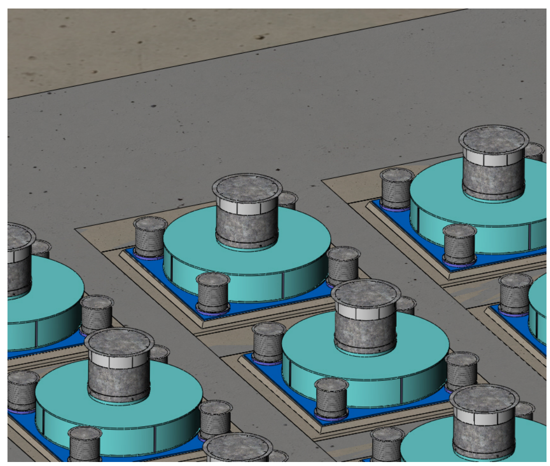
I.  **Reconnect the HI-TRAC to the vertical cask transporter.** **Close the mating device.** **Reinstall the HI-TRAC bottom lid.**	**J.**  **Remove the vertical cask transporter from the mating device.** **Remove the MPC lid rigging from the MPC lid.**
K.  **Remove the mating device from the VVM.**	**L.**  **Reinstall the VVM lid flue extensions.**

FIGURE 9.2.4: PICTORIAL OVERVIEW OF THE LOADING STEPS
(SHEET 3 OF 3)

9.3 ACTIVITIES PERTAINING TO ISFSI OPERATIONS

The HI-STORM UMAX System heat removal is by totally passive means. As discussed in Chapter 10, surveillance of the HI-STORM VVM assembly to ensure its continued effectiveness involves the following principal activities:

1. Check for intrusion of foreign objects that may impair the system's thermal performance during normal operations and in the wake of an extreme environmental phenomenon including severe flood.
2. Check for corrosion damage to the steel parts, namely the CECs (oldest or most vulnerable VVM shall be inspected).
3. Check for structural damage to the ISFSI after an earthquake.
4. Perform the heat removal operability surveillance as specified the CoC.
5. Perform ISFSI Security Operations in accordance with the host site's security plan.

Routine maintenance on the HI-STORM UMAX System will typically be limited to cleaning and touch-up painting of the exposed steel surfaces, repair, and replacement of damaged vent screens, and removal of vent blockages (e.g., leaves, debris), if any (See Table 10.4.1 and 10.4.2 for specific requirements). The heat removal system operability surveillance should be performed after any event that may have an impact on the safe functioning of the HI-STORM UMAX system. These include, but are not limited to, wind storms, heavy snow storms, fire inside the ISFSI, seismic activity, flooding of the ISFSI, and/or observed animal, bird, or insect infestations. The responses to these conditions involve first assessing the dose impact to perform the corrective action (inspect the HI-STORM VVM cavity, clear the debris, check for any structural damage of the ISFSI pad, and/or replace damaged vent screens); perform the corrective action; and verify that the system is operable (check ventilation flow paths and radiation blockage capability). In the unlikely event of significant damage to the ISFSI, possibly from a Beyond-the-Design Basis earthquake, the situation may warrant removal of the MPC, and repair or replacement of the damaged ISFSI areas. Section 9.4 may be used as guidance for unloading the MPC from the HI-STORM UMAX.

9.4 PROCEDURE FOR REMOVING THE MPC FROM THE HI-STORM UMAX VVM CAVITY

9.4.1 Overview of HI-STORM UMAX System Unloading Operations

The MPC is recovered from the HI-STORM UMAX VVM at the ISFSI using the same set of steps as described in Section 9.2, except that the order is basically reversed. The VVM temperature monitoring elements (if used) and lid are removed. The flue extensions are removed as necessary to allow the VCT to access the target cavity. The Mating Device is installed and the Mating Device drawer is partially opened. The MPC lift rigging is attached to the MPC. The MPC rigging is attached to the MPC lift device and positioned on the MPC lid. The Mating Device drawer is closed and the HI-TRAC is positioned on top of the Mating Device and VVM. The HI-TRAC's Bottom lid is unbolted from the HI-TRAC and the Mating Device drawer is opened. The MPC slings are brought through the HI-TRAC and connected to the lift device. The MPC is raised into HI-TRAC and the Mating Device drawer is closed. The bottom lid is bolted to the HI-TRAC. The HI-TRAC is removed from on top of the VVM and transported out of the ISFSI area for further processing. The scope of this FSAR ends with the loaded HI-TRAC suspended from the VCT for further processing. Chapter 9 of the HI-STORM FW System FSAR [9.6.1] for the host MPC provides the direction on further action such as return to the fuel pool or loading in a transport cask for off-site shipment.

9.4.2 MPC Recovery from the HI-STORM UMAX VVM

PRINCIPAL OPERATING STEPS

1. If necessary, perform a transport route walkdown to ensure that the cask transport conditions are met for transporting the loaded HI-TRAC transfer cask. Remove all physical obstructions (e.g., flue extensions) that may interfere with the movement of the VCT.

2. Perform a HI-TRAC receipt inspection and cleanliness inspection in accordance with a written inspection checklist in accordance with the HI-STORM FW System FSAR [9.6.1]. Transport the HI-TRAC to the ISFSI using the cask transporter or other suitable device.

3. Remove the VVM temperature monitoring equipment (if used) from the (MPC donor) VVM cavity.

4. Remove the VVM Closure lid from the donor VVM cavity, preferably keeping its height above the top of the CEC Flange to under 2 feet. See Figure 9.2.3 for a rigging illustration.

Caution:
Operations steps that occur with the MPC in the VVM with the Mating Device installed and the drawer closed must be performed in an expeditious manner to avoid excessive heating of the MPC and fuel. The Mating Device must be removed or the drawer opened to establish air cooling within the time limits described in Section 4.5. In the event of equipment malfunction that results in the blockage of air flow, corrective actions must occur within the time limits of the 100% blocked duct accident condition.

5. Install the Mating Device on the VVM.
6. Partially open the Mating Device drawer.
7. Install the MPC rigging on the MPC lid..
8. Close the Mating Device drawer.
9. If previously drained, fill the neutron shield jacket with plant demineralized water or an approved antifreeze solution as necessary. Ensure that the fill and drain plugs are installed.
10. Align HI-TRAC over the Mating Device and VVM and mate the casks.
11. Unbolt the bottom lid from the HI-TRAC and lower into the Mating Device.
12. Open the Mating Device drawer.
13. At the user's discretion, install temporary shielding to cover the gap above and below the Mating Device drawer.
14. Raise the MPC rigging up through the HI-TRAC and attach them to the lifting device.
15. Raise the MPC into HI-TRAC.
16. Verify the MPC is in the full-up position.
17. Close the Mating Device drawer.
18. Reinstall the bottom lid to the HI-TRAC.
19. Lower the MPC onto the bottom lid.
20. Disconnect the slings from the lifting device and the MPC lift cleats.
21. Remove HI-TRAC from the top of the VVM.

22. Transport the HI-TRAC to the designated location using the cask transporter or other suitable device.

23. Remove the Mating Device from the VVM.

24. Install the VVM lid and vent flue assemblies on all storage cavities, where they were removed, to prevent entry of foreign objects into the VVM.

9.5 REGULATORY COMPLIANCE:

The operational steps required to place a loaded MPC into a HI-STORM UMAX VVM cavity have been described in this chapter. The steps to remove an MPC from a loaded VVM, which are essentially reverse of the steps in the loading sequence, have been provided in Chapter 9 of the HI-STORM FW System FSAR [9.6.1]. These loading steps are, of necessity, generic in their description and may require adaptation to a specific ISFSI. The implementation steps are nevertheless sufficiently detailed to lead to the conclusion that the guidelines of safety and ALARA set down in NUREG-1536 are fully satisfied. In particular, it can be concluded that:

i. There are no radiation streaming paths from the MPC during its transfer operation.
ii. The Mating Device handling operations occur near grade level thus eliminating the need for ladders/platforms and improving the human factors aspects.
iii. There are no freestanding structures in the MPC transfer operations and thus there is no risk of uncontrolled load movement under a (hypothetical) extreme environmental event such as tornado or high winds.
iv. The ventilation paths to passively cool the canister using ambient air during the transfer operation is maintained at all times (except during brief operations as mentioned above) thus protecting the fuel cladding from overheating and eliminating any thermally guided time limit on the duration for implementing the transfer steps.
v. All heavy load handling is carried out by handling devices that are equipped with redundant load drop protection features.
vi. Each storage cavity is independently accessible. Installation or removal of any MPC does not have to contend with other stored MPCs.
vii. Because the MPC insertion (and withdrawal) occurs in the vertical configuration with ample lateral clearances, there is no risk of scratching or gouging of the MPC's external surface (Confinement Boundary). Thus the ASME Section III Class 1 prohibition against damage to the pressure retaining boundary is maintained.

It is thus concluded that the HI-STORM UMAX ISFSI is engineered to meet the safety and ALARA imperatives contemplated in 10CFR 72 in full measures.

9.6 REFERENCES:

[9.6.1] "Final Safety Analysis Report on the HI-STORM FW System", Holtec Report No. HI-2114830, latest revision.

CHAPTER 10: ACCEPTANCE CRITERIA AND MAINTENANCE PROGRAM

10.0 INTRODUCTION

This chapter addresses the fabrication, inspection, test, and maintenance programs for the HI-STORM UMAX VVM assemblies. In particular, this chapter identifies the fabrication, inspection, test, and maintenance programs to be conducted on the HI-STORM UMAX VVM to verify that the structures, systems, and components (SSCs) classified as important-to-safety have been fabricated, assembled, inspected, tested, accepted, and maintained in accordance with the requirements set forth in this FSAR, the applicable regulatory requirements, and the Certificate of Compliance (CoC). The acceptance criteria and maintenance program requirements specified in this chapter apply to each HI-STORM UMAX system fabricated, assembled, inspected, tested, and accepted for use under the purview of the system's CoC. The assessment of the fabrication, inspection, test, and maintenance programs for the MPC and HI-TRAC used with the HI-STORM UMAX is described in Chapter 10 of the HI-STORM FW FSAR, a QA validated copy of which is placed in this docket for reference. The material in the HI-STORM FW FSAR which is relied upon to articulate the acceptance criteria and maintenance program of components that the "UMAX" system shares with "FW" is provided in a matrix form in Table 10.0.1 for ease of reference.

The controls, inspections, and tests set forth in this chapter, in conjunction with the design requirements described in previous chapters, ensure that the HI-STORM UMAX system will reject the decay heat of the stored radioactive materials in the MPCs to the environment by passive means and maintain radiation doses within regulatory limits. The controls, inspections, and tests set forth in Chapter 10 of the HI-STORM FW along with the MPC and HI-TRAC design requirements described in the HI-STORM FW, ensure that the system will maintain confinement of radioactive material and will maintain subcriticality control under normal, off-normal, and hypothetical accident conditions, including short term operations. The aforementioned design, controls, inspections, and tests described in the HI-STORM FW also ensure that the MPC and HI-TRAC will reject the decay heat of the stored radioactive materials to the environment by passive means and maintain radiation doses within regulatory limits.

Both pre-operational and operational tests and inspections are performed throughout HI-STORM UMAX system operations to assure that the system is functioning within its design parameters. These include receipt inspections, nondestructive weld examinations, thermal performance tests, and others. "Pre-operation", as referred to in this chapter, defines that period of time from acceptance inspection of a HI-STORM UMAX system until the loaded MPC is placed in the CEC cavity.

The HI-STORM UMAX system is classified as important-to-safety. Therefore, the individual structures, systems, and components (SSCs) that make up the HI-STORM UMAX system shall be designed, fabricated, assembled, inspected, tested, accepted, and maintained in accordance

with a quality program commensurate with the particular SSC's graded quality category. The licensing drawings identify all important-to-safety subcomponents of the HI-STORM UMAX system.

The acceptance test requirements on the manufactured welds in the HI-STORM UMAX system are contained in the component licensing drawings in Section 1.5. Additional details on the requirements in the drawings are provided in this chapter or in Chapter 10 of the HI-STORM FW FSAR for MPC and HI-TRAC, which will be incorporated in the shop manufacturing documents (viz., weld procedures, shop travelers, inspection procedures, and fabrication procedures) to ensure full compliance with this FSAR.

The VVM consists of a shop-fabricated CEC (Cavity Enclosure Container) installed below grade and a removable Closure Lid. The CEC is a welded shell-type structure made of steel plate and bar (or forging) stock. Likewise, the Closure Lid is made of welded and formed steel plates and bar (or forging) stock. However, unlike the CEC, the Closure Lid also contains shielding concrete.

By virtue of its underground configuration, the CEC is interfaced by the subgrade along its lateral surface, by the ISFSI pad near its flanged upper region and by the Support Foundation Pad (SFP) along its bottom surface. The requirements on these interfacing bodies, to the extent they are needed to enable the CEC to render its intended function, are provided in Chapter 2. All requirements pertaining to the manufacturing, inspection, testing, and maintenance of the VVM SSCs are presented in this chapter to comply with the provisions of 10CFR72.24(p).

The user of the HI-STORM UMAX system should consult this chapter and the Technical Specification to ensure that the site's acceptance criteria and maintenance program are consistent with the representations made herein. The user's maintenance program should also utilize the information that will be gathered and disseminated by Holtec International through the Company's "lessons learned" data base and other means that are intended to help ensure maximum shielding effectiveness and a long service life of the ISFSI .

TABLE 10.0.1: APPLICABLE SECTIONS OF HI-STORM FW FSAR[1]

Location of UMAX FSAR	**Subject of the Reference**	**Location in Chapter 10 of the HI-STORM FW FSAR, Revision 3**
Section 10.0	Fabrication, inspection, test, and maintenance programs for the MPC and HI-TRAC	Table 10.2.1 for maintenance for HI-TRAC Sub-Sections 10.2.1, 10.2.4, and 10.2.5
Section 10.1	Inspection and acceptance tests for MPCs and HI-TRAC	Section 10.1
Section 10.1	The testing and inspection acceptance criteria applicable to the MPC, including the MPC Lid-to-Shell weld, and the HI-TRAC	Table 10.1.1 and 10.1.3
Subsection 10.1.1	Fabrication controls and required inspections	Sub-Section 10.1.1
Subsection 10.1.1	Weld examination requirements for the MPCs and HI-TRAC	Sub-Section 10.1.1;.4 Tables 10.1.4 and 10.1.5
Sub-Section 10.1.2	Structural and pressure test requirements for the MPC and HI-TRAC	Sub-Section 10.1.2
Sub-Section 10.1.3	Material testing requirements for the MPC (except Metamic-HT) and HI-TRAC	Sub-Section 10.1.3
Sub-Section 10.1.3	Metamic-HT testing requirements	Sub-Section 10.1.3 of latest implemented revision of HI-STORM FW FSAR and PS-11 [10.1.6]
Sub-Section 10.1.4	Leakage testing of MPC	Sub-Section 10.1.4
Subsection 10.1.5	Requirements for component tests of valves, pressure relief device and fluid transport devices associated with the MPC and HI-TRAC	Sub-Section 10.1.5
Sub-Section 10.1.6	The shielding integrity testing requirements for the MPC and HI-TRAC	Sub-Section 10.1.6
Sub-Section 10.1.7	The thermal test requirement for the first manufactured MPC, either MPC-37 or MPC-89	Sub-Section 10.1.7

[1] For convenience of reference, the specific revision of the HI-STORM FW FSAR that is referenced in the safety analysis herein is placed in this docket. Updated versions of the HI-STORM FW FSAR shall be placed in this docket as necessary so as to ensure that the safety analyses on the “UMAX” docket (72-1040) remain aligned with the material referenced in the HI-STORM FW FSAR.

10.1 ACCEPTANCE CRITERIA

This section provides the workmanship inspections and acceptance tests to be performed on the HI-STORM UMAX ISFSI and VVM prior to and during loading of the system. Information on the workmanship inspections and acceptance tests to be performed on the MPC and HI-TRAC components are provided in Section 10.1 of the HI-STORM FW FSAR. These inspections and tests provide the assurance that all components of the HI-STORM UMAX system are fabricated, assembled, inspected, tested, and accepted for use under the conditions specified in this FSAR and the Certificate of Compliance issued by the NRC in accordance with the provisions of 10CFR72.

The testing and inspection acceptance criteria applicable to HI-STORM UMAX VVM are listed in Table 10.1.1 and discussed in more detail in the sections that follow. The testing and inspection acceptance criteria applicable to the MPC, including the MPC Lid-to-Shell weld, and the HI-TRAC are listed in Table 10.1.1 and 10.1.3 of the HI-STORM FW FSAR and are discussed in more detail in the HI-STORM FW FSAR. Chapter 9 from both this FSAR and the HI-STORM FW FSAR provide operating guidance. Chapter 13 of this FSAR provides the bases for the Technical Specifications. These inspections and tests are intended to demonstrate that the HI-STORM UMAX system has been fabricated, assembled, and examined in accordance with the design criteria contained in Chapter 2 of the HI-STORM UMAX FSAR. Identification and resolution of manufacturing non-compliances, if any, shall be performed in accordance with the Holtec International Quality Assurance Program approved by the USNRC (Docket Number 71-0784).

The material on testing and maintenance of system components in this FSAR governs the content of the daughter documents such as the Manufacturing Manual and the O&M Manual for the system components used in the manufacturing and long-term maintenance of the system components.

10.1.1 Fabrication and Nondestructive Examination (NDE)

This subsection summarizes the test program required for the HI-STORM UMAX system.

10.1.1.1 Fabrication Requirements

The manufacturing of UMAX VVM components shall be carried out in accordance with the CoC holder's NRC-approved QA program. All elements of the manufacturing cycle will be established to accord with the Important-to-Safety (ITS) designation of the specific part (indicated in the Licensing drawings) and the applicable provisions of the referenced codes and standards. The acceptance criteria for the manufactured components apply to each step of the manufacturing evolution, namely (a) supplier selection, (b) preparation of material procurement specifications, (c) preparation of the shop traveler and fabrication procedures, (d) fabrication activities such as forming, bending, plasma cutting, and welding, (e) in-process inspections, (f)

final inspection, (g) packaging for shipment, and (h) assembling of the documentation package to serve as the archival evidence of adherence to the quality requirements.

In order to receive the Certificate-of-Compliance under the CoC holder's QA program, the manufacturing of the HI-STORM UMAX VVM components must meet all of the technical, quality control, procedural (quality assurance) and administrative requirements set forth in the manufacturing program.

Similarly, the steel plates, bars, and forgings, as applicable, used in the construction of the HI-STORM UMAX VVM shall be dimensionally inspected to assure compliance with the requirements on the drawings. Test results shall be documented and become part of the quality documentation package. Dimensional inspections of the Closure Lid and its weight measurement after placement of the shielding concrete shall assure that the required amount of shielding material has been incorporated in the lid.

The fabrication controls and required inspections detailed in Section 10.1 of the HI-STORM FW FSAR shall be performed on the MPCs and HI-TRAC transfer casks, in order to assure compliance with the HI-STORM FW FSAR, the HI-STORM UMAX FSAR and the Certificate of Compliance. Specifically, the following fabrication controls and required inspections shall be performed on the HI-STORM UMAX system in order to assure compliance with this FSAR and the Certificate of Compliance.

i. Materials of construction specified for the HI-STORM UMAX system are identified in the drawings in Chapter 1. Materials for ITS Components shall be procured with certification and supporting documentation as required by the ASME Code, Section II, where applicable [10.1.1], Holtec procurement specifications, and 10CFR72, Subpart G. ITS materials and components shall be receipt inspected for visual and dimensional acceptability, material conformance to specification requirements, and traceability markings, as applicable. Controls shall be in place to ensure that material traceability is maintained throughout fabrication.

ii. Structural welds shall be performed using welders and weld procedures that have been qualified in accordance with ASME Code Section IX and ASME Section III, Subsection NF. Non-structural welds shall be performed using welders and weld procedures that have been qualified in accordance with ASME Code Section IX or AWS D1.1.

iii. Structural welds shall be visually examined in accordance with ASME Code, Section V, Article 9. NDE inspections of structural welds shall be performed in accordance with written and approved procedures by personnel qualified in accordance with SNT-TC-1A [10.1.2] or other site-specific, NRC-approved program for personnel qualification. Non-structural welds shall be visually examined by qualified welders or weld inspectors for cracks and other linear indications which must be repaired when identified.

iv. Single pass welds identified on the HI-STORM UMAX VVM drawing shall be made using a prequalified weld procedure. In addition to the normal procedure qualifications required by ASME Section IX or AWS D1.1, the weld procedure shall define limits on essential variables such that the weld produced will be guaranteed to have a minimum fillet size of 1/8”. Upon qualification of the weld procedure, the single pass welds identified on the licensing drawing can be fabricated and verification of use of proper essential variables for welding can be substituted for examination for proper weld size. Single pass welds shall be visually examined by qualified welders or weld inspectors for cracks and other linear indications which must be repaired when identified.

v. The HI-STORM UMAX system shall be inspected for cleanliness and proper packaging for shipping in accordance with written and approved procedures.

vi. Each VVM shall be durably marked with the appropriate model number, a unique identification number, and its empty weight per 10CFR72.236(k) at the completion of the acceptance test program.

vii. A documentation package shall be prepared and maintained during fabrication of each HI-STORM UMAX system to include detailed records and evidence that the required inspections and tests have been performed. The completed documentation package shall be reviewed to verify that the HI-STORM UMAX system components have been properly fabricated and inspected in accordance with the design and Code construction requirements. The documentation package shall include, as applicable, but not be limited to:

- Completed Shop Weld Records
- Inspection Records
- Nonconformance Reports
- Material Test Reports
- NDE Reports
- Dimensional Inspection Report

10.1.1.2 Visual Inspections and Measurements

The HI-STORM UMAX system components shall be assembled in accordance with the licensing drawing package in Section 1.5. The drawings provide dimensional tolerances that define the limits on the dimensions used in licensing basis analysis. Fabrication drawings provide additional dimensional tolerances necessary to ensure fit-up of parts. Visual inspections and measurements shall be made and controls shall be exercised to ensure that the cask components conform to the dimensions and tolerances specified on the licensing and fabrication drawings. These dimensions are subject to independent confirmation and documentation in accordance with the Holtec QA program approved in NRC Docket Number 71-0784.

The HI-STORM UMAX VVM, except for its Closure Lid, does not require unique marking other than that required for fuel ISFSI configuration management because it is not a transportable structure. The module is essentially an integral portion of the ISFSI, not a separate cask structure that can be moved by the user. As such, the markings provided on the MPC alone are sufficient to meet the requirements of 10CFR72.236(k). Nevertheless, it is required that each VVM cavity be labeled with an identifier that is unique for the specific site. Further, because the Closure Lid will be subject to handling, consistent with defense-in-depth guidelines for handling heavy loads at nuclear plants, its unique identifier and its bounding weight shall be permanently marked on a readily visible location.

The following shall be verified as part of visual inspections and measurements:

- Visual inspections and measurements shall be made to ensure that the systems' effectiveness is not significantly reduced as a result of manufacturing deviations. Any *important-to-safety* component found to be under the specified minimum thickness shall be justified under the rules of 10CFR72.48 or repaired or replaced, as appropriate.

- The HI-STORM UMAX CEC and Closure Lid shall be inspected to confirm that the labeling is complete and legible.

- The system components shall be inspected for cleanliness and prepared for shipment in accordance with written and approved procedures.

- Visual inspections shall be made to verify that neutron absorber panels and basket shims are present as required by the MPC basket design.

The visual inspection and measurement results for the HI-STORM UMAX system shall become part of the final quality documentation package.

10.1.1.3 Weld Examination

The examination of the HI-STORM UMAX system welds shall be performed in accordance with the drawing package in Section 1.5 and the applicable codes and standards.

Weld examination requirements for the MPC and HI-TRAC are described in Section 10.1 of the HI-STORM FW FSAR. All structural weld inspections shall be performed in accordance with written and approved procedures by personnel qualified in accordance with SNT-TC-1A. Structural welds for the VVM shall meet the acceptance criteria of ASME Section III, Subsection NF. All required inspections, examinations, and tests shall become part of the final quality documentation package. Non-structural weld inspections shall be performed by welders or weld inspectors qualified in accordance with written procedures. Non-structural welds shall be free of cracks and other linear indications.

10.1.2 Structural and Pressure Tests

The structural and pressure test requirements for the MPC and HI-TRAC are described in Section 10.1 of the HI-STORM FW FSAR. The following subparagraphs provide the structural and pressure test requirements for the HI-STORM UMAX VVM.

10.1.2.1 Lifting Locations

Because the HI-STORM UMAX system is immovable, it does not utilize any lifting appurtenances.

The only removable component in the HI-STORM UMAX system is the Closure Lid which features four lift lugs (see drawing in Section 1.5). Because the lugs are integral to the component, they possess high ductility and, as shown in Chapter 3, meet the factor of safety of 6 to yield and 10 to ultimate, as required by ANSI N14.6 [10.1.3]. The requirements for testing and inspection of the load handling equipment related to the HI-TRAC and MPC are covered in the governing license document for the respective component and are not repeated here.

Section 5 of NUREG-0612 [10.1.4] calls for measures to "provide an adequate defense-in-depth for handling of heavy loads...". The NUREG-0612 guidelines cite four major causes of load handling accidents, of which rigging failure is one:

i. operator errors
ii. rigging failure
iii. lack of adequate inspection
iv. inadequate procedures

The cask loading and handling operations program shall ensure maximum emphasis to mitigate the potential load drop accidents by implementing measures to eliminate shortcomings in all aspects of the operation including the four aforementioned areas.

Each lifting lug will be subjected to a dimensional test in the shop to ensure that it meet the dimensional requirements. The lift lugs will be load tested per ANSI 14.6 -1993 prior to the shipping of the Closure Lid to the ISFSI site.

10.1.3 Materials Testing

The materials testing requirements for the MPC and HI-TRAC are described in Section 10.1 of the HI-STORM FW FSAR. The following paragraphs provide the materials testing requirements for the HI-STORM UMAX VVM.

The steel materials used in the HI-STORM VVM will be tested in accordance with the requirements of the applicable material code.

The concrete utilized in the construction of the HI-STORM UMAX ISFSI shall be mixed, poured, and tested in accordance with the applicable code using written and approved procedures. Testing shall verify the material properties meet design requirements. Tests required shall be performed at a frequency as defined in the applicable ACI code.

10.1.4 Leakage Testing

Leakage testing shall be performed in accordance with the requirements of ANSI N14.5 [10.1.5]. Testing shall be performed in accordance with written and approved procedures with acceptance criteria being "leak tight" as defined in ANSI N14.5.

Helium leakage testing of the MPC base metals (shell, baseplate, and MPC lid) and MPC shell to baseplate and shell to shell welds is performed on the unloaded MPC. The helium leakage test of the vent and drain port cover plates and cover plate to MPC lid welds (field welds) shall be performed using procedures meeting the requirements of ANSI N14.5.

If a leakage rate exceeding the acceptance criterion is detected, then the cause of the leakage shall be determined and the area repaired per the requirements of ASME Code Section III, Subsection NB, Article NB-4450. Re-testing shall be performed until the leakage rate acceptance criteria is met.

Leak testing results for the MPC shall be documented and shall become part of the quality record documentation package.

There is no leakage testing required for the HI-STORM UMAX VVM.

10.1.5 Component Tests

10.1.5.1 Valves, Pressure Relief Devices, and Fluid Transport Devices

There are no valves, pressure relief device and fluid transport devices associated with the HI-STORM UMAX VVM. The requirements for component tests of valves, pressure relief device and fluid transport devices associated with the MPC and HI-TRAC are described in Section 10.1 of the HI-STORM FW FSAR.

10.1.5.2 Seals and Gaskets

There are no confinement seals or gaskets included in the HI-STORM UMAX system.

10.1.6 Shielding Integrity

The shielding integrity testing requirements for the MPC and HI-TRAC are described in Section 10.1 of the HI-STORM FW FSAR. There is no criticality control testing required for the HI-STORM UMAX VVM. The shielding integrity testing for the HI-STORM UMAX VVM is limited to dimensional checks of the steel components and concrete forming and to confirmation of the correct density for the concrete used in the Closure Lid and ISFSI pad.

10.1.7 Thermal Acceptance Tests

The thermal performance of the HI-STORM UMAX system, including the MPCs and HI-TRAC transfer cask, is demonstrated through analysis in Chapter 4 of this FSAR and the HI-STORM FW FSAR. Dimensional inspections to verify the item has been fabricated to the dimensions provided in the drawings shall be performed prior to system loading.

The thermal test requirement for the first manufactured MPC, either MPC-37 or MPC-89 is described in Section 10.1 of the HI-STORM FW FSAR.

Table 10.1.1

HI-STORM UMAX VVM ASSEMBLY INSPECTION AND TEST ACCEPTANCE CRITERIA

Function	**Fabrication**	**Pre-operation**	**Maintenance and Operations**
Visual Inspection and Nondestructive Examination (NDE)	Structural Steel Components: a) All structural welds shall be visually examined per ASME Section V, Article 9 with acceptance criteria per ASME Section III, Subsection NF, NF-5360. b) All structural welds requiring MT examination as shown on the drawings shall be MT examined per ASME Section V, Article 7 with acceptance criteria per ASME Section III, Subsection NF, NF-5340. c) NDE of weldments shall be defined on design drawings using ANSI NDE symbols and/or notations.	a) The VVM shall be visually inspected for general condition of coatings and insulation and for presence of FME prior to placement in service. b) VVM Assembly protection at the licensee's facility shall be verified. c) Exclusion of foreign material shall be verified prior to placing the VVM Assembly in service at the licensee's facility.	a) Indications identified during visual inspection shall be corrected, reconciled, or otherwise dispositioned. b) Exposed surfaces shall be monitored for coating deterioration and repair/recoat as necessary.

Table 10.1.1 (continued)

HI-STORM UMAX VVM ASSEMBLY INSPECTION AND TEST ACCEPTANCE CRITERIA

Function	**Fabrication**	**Pre-operation**	**Maintenance and Operations**
Visual Inspection and Nondestructive Examination (NDE) (continued)	General: a) Cleanliness of the VVM Assembly shall be verified upon completion of fabrication. b) Packaging of the VVM Assembly at the completion of shop fabrication shall be verified prior to shipment. c) Labeling of the CEC and the Closure Lid	General: a) Labeling of the CEC and the Closure Lid	
Structural	a) Verification of structural materials shall be performed through receipt inspection and review of certified material test reports (CMTRs) obtained in accordance with the item's quality category. b) Load testing of Closure Lid lift points	a) No structural or pressure tests are required for the VVM during pre-operation.	a) No structural or pressure tests are required for the VVM during operation.
Leak Tests	a) None.	a) None.	a) None.
Shielding Integrity	a) Concrete density shall be verified per Appendix 1.D of [10.1.4], at time of placement. b) Shell thicknesses and dimensions between inner and outer shells shall be verified as conforming to design drawings prior to concrete placement in the Closure Lid.	a) None	a) None.

Table 10.1.1 (continued)

HI-STORM UMAX VVM ASSEMBLY INSPECTION AND TEST ACCEPTANCE CRITERIA

Function	**Fabrication**	**Pre-operation**	**Maintenance and Operations**
Thermal Acceptance	a) Inner shell I.D. and vent size, configuration and placement shall be verified.	a) Visual examination of the insulation and coatings will be performed to verify that they are in good condition b) Air flow paths will be visually inspected to ensure they are clear and free of FME.	a) MPC lid temperature test shall be performed as discussed in Section 10.3 b) Periodic surveillance shall be performed by either (1) or (2) below, at the licensee's option. (1) Inspection of VVM inlet and outlet air vent openings for debris and other obstructions. (2) Temperature monitoring.
Cask Identification	a) Verification that the VVM identification is present in accordance with the drawings shall be performed upon completion of assembly.	a) The VVM identification shall be checked prior to loading.	a) The VVM identification shall be periodically inspected per licensee procedures and repaired or replaced if damaged.
Fit-up Tests	a) Where closely toleranced alignment between mating components is required, a component-to-component fit-up shall be performed directly whenever practical or using templates or other means.	a) Lid fit-up with the VVM shall be verified following placement of concrete around the VVM.	a) None.

10.2 SITE CONSTRUCTION

Like the aboveground HI-STORM overpacks, the site construction activities on the HI-STORM UMAX VVM components and ISFSI structures (namely, the Support Foundation Pad, the ISFSI Pad Subgrade, the ISFSI Pad, and the Enclosure Wall) shall be carried out to demonstrate compliance with the technical criteria set forth in this FSAR. The specific requirements, to ensure that the required *critical characteristics** of the VVM and the interfacing SSCs are realized, are summarized below.

a. All site construction processes shall be controlled by procedures, work instructions, or similar processes. These shall be reviewed and approved in accordance with the CoC holder or the site-licensee's NRC approved QA program.

b. All HI-STORM UMAX VVM components and structures designated as ITS shall be subject to the necessary quality assurance regimen established in accordance with the Company's QA program. The major NITS components in the ISFSI are the inlet and outlet flue extensions including the chimney cover and the concrete casement around the CEC shell, and the cathodic protection system (if used).

c. Compliance with the requirements in this FSAR shall be demonstrated by appropriate testing and the results documented for archival reference. For example, the strength properties of the subgrade can be established using the classical "plate test" or an equivalent method endorsed by a national consensus standard.

d. The density and compressive strength of the ISFSI concrete and the yield strength of the re-bars used in the ISFSI structures shall be confirmed to comply with the Purchasing Specification prepared by the cask designer for the particular site.

e. The insulation installed on the Divider shell shall be subject to visual examination to ensure that it is undamaged and properly secured.

f. The coating on the exterior surface, if used, of the CEC shall be inspected for absence of gross damage before the surface becomes inaccessible.

g. The concrete encasement, if used, shall be installed in accordance with the provisions of Chapter 8.

h. The dimensional compliance of the CEC, including its verticality, shall be inspected to establish compliance with the governing construction documents.

i. The installation of the corrosion barriers shall be in accordance with written procedures.

* See Glossary for definition.

j. All VVM surfaces that become inaccessible shall be photographed with sufficient resolution to provide a clear archive of their in-situ state at installation.

10.3 INSPECTIONS AND TESTING

i. Post-Construction Inspection:

Each as-built HI-STORM UMAX VVM shall be inspected for final acceptance before it is loaded with fuel. The following inspections define minimum acceptance requirements:

a. The as-installed CEC shall be inspected to ensure that it will not hinder installation of the Closure Lid.

b. The CEC Flange Shell gasket/seal bearing surfaces shall be inspected for its horizontal alignment (within specified tolerance). The seals shall be inspected for general condition (lack of cuts, tears, or degradation that could lead to poor sealing).

c. The Closure Lid skirt shall be checked for fit-up with the CEC Flange.

d. The outlet air passage in the Closure Lid shall be inspected for absence of obstruction such as debris and extensive weld spatter.

e. The results of the post-construction inspection shall be incorporated in the VVM's Documentation Package.

f. The impressed current cathodic protection system (ICCPS), if used, shall be tested for operability using Holtec provided procedures.

ii. Shielding Integrity and Effectiveness Test:

Operational neutron and gamma shielding effectiveness tests shall be performed after the first fuel loading at the host plant site using written and approved procedures. Calibrated neutron and gamma dose rate meters shall be used to measure the actual neutron and gamma dose rates at the accessible surface of the HI-STORM UMAX VVM. Measurements shall be taken at the locations specified in the Radiation Protection Program for comparison against the prescribed limits. The test is performed to identify the expected dose levels around the VVM in order to plan for appropriate radiation protection measures for future cask loadings. Dose rate measurements shall be documented and shall become part of the quality record of the loaded cask.

iii. Thermal Acceptance Test

The thermal performance of the HI-STORM UMAX system is demonstrated through analysis in Chapter 4 of this FSAR.

However, a thermal acceptance test shall be performed on the first fully loaded VVM assembly whose aggregate MPC heat load is at least 50% of the Design Basis maximum heat load per the system CoC. (Because of its in-ground installation, a thermal test at a lower heat load is apt to be too inaccurate to provide meaningful information.)

After the system has been in storage for at least one week (to reach steady state) and when the ambient is relatively quiescent and the solar heat deposition rate is minimal, the surface temperature of the top of the MPC lid shall be measured at the center and 4 orthogonally disposed peripheral locations.

The measured lid temperature data (at designated measurement points) shall be compared with the results predicted by the FLUENT analysis.

The measured temperatures (in Deg. F) must be no greater than 5% over the corresponding analytically predicted temperatures to be acceptable. Otherwise, the Design Basis heat load capacity of the VVM assembly will be downgraded by incorporating a penalty factor in the FLUENT model and the USNRC so informed. The reduced heat load values will become the *de facto* limits for all VVMs at the site until a root cause evaluation indicates extenuating circumstances unique to the site.

The heat load anomaly must be resolved by additional testing: until then, the reduced heat load limit will apply to all HI-STORM UMAX sites.

The technical specifications require periodic surveillance of the system air inlet and outlet vents or, optionally, implementation of a HI-STORM UMAX VVM air temperature monitoring program to provide continued assurance of the operability of the HI-STORM UMAX heat removal system.

iv. Storm Water Control Test

The HI-STORM UMAX VVM is designed to direct storm water and snow/ice melt-off away from the CEC Flange and the Closure Lid where the air passages are located. The engineered rain caps installed on the inlet and outlet serve to keep rain and snow away from the VVM cavity. Moreover, any minor amount of moisture that may intrude into the MPC storage cavity due to wind-driven rain will evaporate in a short period of time due to the continuous movement of heated air in the MPC storage cavity. To verify the effectiveness of the storm water drainage design, a one-time test shall be performed after construction of the first VVM to ensure that the design is effective in directing storm water away from the VVM to the ISFSI's drainage system. The VVM shall be subjected to a water spray that simulates exposure to rainfall of at least 2

inches per hour for at least one hour. At the conclusion of the water spray, the depth of the water (if any) in the bottom of the module cavity shall be measured. Any amount of water accumulation beyond wetting of the Bottom Plate indicates an inadequacy in rain diversion features of the VVM and shall be appropriately corrected. It should be noted that accumulation of water is not injurious to the thermal performance of the system. The only deleterious effect of prolonged exposure to water is the potential for reducing the service life of the preservative on the wetted surface of the Bottom Plate, CEC shell, and other interfacing components.

10.4 MAINTENANCE PROGRAM

An ongoing maintenance program shall be defined and incorporated into the HI-STORM UMAX system Operations and Maintenance Manual, which shall be prepared and issued prior to the first use of the system by a user. This document shall delineate the detailed inspections, testing, and parts replacement necessary to ensure continued structural, thermal performance, and radiological safety in accordance with 10CFR72 regulations, the conditions in the Certificate of Compliance, and the design requirements and criteria contained in this FSAR.

This section addresses the maintenance program for the HI-STORM UMAX VVM. The HI-STORM UMAX system does not require any changes to the maintenance requirements for the MPC described in its governing FSARs.

The HI-STORM UMAX system is totally passive by design and requires minimal preventive maintenance to ensure that it will render its intended design functions satisfactorily. Periodic surveillance (via temperature monitoring or visual or camera-aided inspection of air passages) is required to ensure that the air passage in the VVM is not blocked. Preventive or remedial painting of the exposed steel surfaces as part of the user's preventive maintenance program is recommended to mitigate corrosion. Such preventive maintenance activities are typical in scope and complexity to other standard maintenance activities at nuclear power plants.

In-service inspection for long-term interior and below-grade degradation shall be performed on a site-specific basis in accordance with Holtec specified long-term maintenance guidelines and the licensee's preventive maintenance program. At most potential ISFSI sites, visual inspection of accessible areas of the HI-STORM UMAX VVM is expected to be sufficient to detect in-service degradation. The frequency of this visual in-service inspection should be in performed in accordance with Table 10.4.1

Additional in-service inspection activities may include more thorough inspections for foreign material accumulation, corrosion (CEC wall thinning) and insulation degradation as warranted by site-specific conditions. In-service inspections for evaluating foreign material accumulation, corrosion (CEC wall thinning) and/or insulation degradation are not required if it is determined that the applicable degradation actuating mechanisms do not exist. A VVM with a loaded MPC may be inspected using remote devices such as a boroscope. The oldest VVM or VVM considered to be most vulnerable to corrosion degradation shall be selected for inspection.

As is true for all components certified pursuant to this FSAR, the maintenance activities on the HI-STORM UMAX VVM shall be performed in accordance with a written program that fulfills the requirements of the CoC holder's 10CFR72 Subpart G compliant QA program; the owner site's Safety Plan and corrective action program; and the system's Technical Specification.

Among the QA commitments are performance of maintenance by trained personnel by written procedures and written documentation of the maintenance work performed and of the results

obtained. Table 10.4.2 provides a listing of the minimum maintenance activities on the HI-STORM UMAX VVM.

In summary, the HI-STORM UMAX System is totally passive by design: There are no active components or monitoring systems required to assure the performance of its safety functions. As a result, only minimal maintenance will be required over its lifetime, and this maintenance would primarily result from the effects of weather. Typical of such maintenance would be the reapplication of corrosion inhibiting materials on accessible external surfaces. Visual inspection of the vent screens is required to ensure the air flow passages are free from obstruction. Such maintenance requires methods and procedures that are far less demanding than those currently in use at power plants.

Maintenance activities shall be performed under the licensee's NRC-approved quality assurance program. Maintenance activities shall be administratively controlled and the results documented.

10.4.1 Structural and Pressure Parts

Prior to each MPC loading, a visual examination in accordance with a written procedure shall be required of the Closure Lid lift lugs and the HI-TRAC Tapped Anchor Locations (TALs), bottom lid bolts, and bolt holes. The examination shall inspect for indications of overstress such as cracks, deformation, wear marks, corrosion, etc. Repairs in accordance with written and approved procedures shall be required if an unacceptable condition is identified.
As described in Chapters 7 and 12 of this FSAR, there are no credible normal, off-normal, or accident events which can cause the structural failure of the MPC. Therefore, periodic structural or pressure tests on the MPCs following the initial acceptance tests are not required as part of the storage maintenance program.

10.4.2 Leakage Tests

There are no seals or gaskets used on the fully-welded MPC confinement system. As described in Chapters 7 and 12, there are no credible normal, off-normal, or accident events which can cause the failure of the MPC Confinement Boundary welds. Therefore, leakage tests are not required as part of the storage maintenance program.

10.4.3 Subsystem Maintenance

The HI-STORM UMAX System does not include any active subsystems that provide auxiliary cooling. If the cask user chooses to use an air temperature monitoring system in lieu of visual inspection of the air inlet and outlet vents, the thermocouples and associated temperature monitoring instrumentation shall be maintained and calibrated in accordance with the user’s QA program commensurate with the equipment’s safety classification and designated QA category.

10.4.4 Pressure Relief Devices

The pressure relief devices used on the water jackets for the HI-TRAC transfer cask shall be confirmed to have been calibrated as specified in the licensing basis FSAR for the HI-TRAC to ensure pressure relief settings are accurate prior to the cask's use at a HI-STORM UMAX ISFSI.

10.4.5 Shielding

The gamma and neutron shielding materials in the HI-TRAC and MPC are not subject to measurable degradation over time or as a result of usage. The radiation shielding capacity of the HI-STORM UMAX System is expected to remain undiminished over time. Therefore, unless the VVM is subjected to an extreme environmental event that imparts stresses or temperatures beyond-the-design-basis limits for the system (i.e., prolonged fire or impact from a beyond-the-design basis large energetic projectile) with the plausible potential to degrade the shielding effectiveness of the VVM, no shielding effectiveness tests beyond that required by the plant's Radiation Protection Program are required over the life of the HI-STORM UMAX System.

Radiation monitoring of the ISFSI by the licensee in accordance with 10CFR72.104(c) provides ongoing evidence and confirmation of shielding integrity and performance. If increased radiation doses are indicated by the facility monitoring program, additional surveys of the ISFSI shall be performed to determine the cause of the increased dose rates.

The water level in the HI-TRAC water jacket shall be verified during each loading campaign in accordance with the licensee's approved operations procedures.

The neutron absorber panels installed in the MPC baskets are not expected to degrade under normal long-term storage conditions. Therefore, no periodic verification testing of neutron poison material is required on the HI-STORM UMAX System.

10.4.6 Thermal

In order to assure that the HI-STORM UMAX System continues to provide effective thermal performance during storage operations, surveillance of the air vents (or alternatively, by temperature monitoring) shall be performed in accordance with written procedures.

For those licensees choosing to implement temperature monitoring as the means to verify VVM Assembly heat transfer system operability, a maintenance and calibration program shall be established in accordance with the plant-specific Quality Assurance Program, the equipment's quality category, and manufacturer's recommendations.

Table 10.4.1

HI-STORM SYSTEM MAINTENANCE PROGRAM SCHEDULE

Task	Frequency
VVM cavity visual inspection	Prior to MPC loading
Divider shell visual inspection	Prior to MPC loading
Closure Lid visual inspection	Prior to MPC loading
VVM external surface (accessible) visual examination	Annually, during storage operation
VVM inlet and outlet vent screen visual inspection for damage, holes, etc.	Monthly
VVM inlet and outlet vent inspection for blockage	Daily unless monitoring is performed using temperature monitoring equipment
HI-TRAC cavity visual inspection	Prior to each handling campaign
HI-TRAC TAL visual inspection	Prior to each handling campaign
HI-TRAC bottom lid bolts and bolt holes	Prior to each handling campaign
HI-TRAC pressure relief device calibration	Per the device manufacturer's recommendation.
HI-TRAC water jacket water level visual examination	During each handling campaign in accordance with licensee approved operations procedures
VVM visual inspection of identification markings	Annually
VVM Air Temperature Monitoring System	Per licensee's QA program and manufacturer's recommendations
VVM inlet plenum inspection for accumulation of FME.	Every five years or following a severe weather event that may introduce significant FME material.

Table 10.4.2

MAINTENANCE ACTIVITIES FOR THE HI-STORM UMAX VVM

	Activity	Frequency	Comment
1.	CEC cavity is visually inspected	Prior to MPC installation	To ensure that VVM internal components are properly aligned, the surface preservatives on all exposed surfaces are undamaged, the insulation on the Divider Shell is undamaged and the cavity is free of visible foreign material.
2.	Lid Examination	Prior to MPC installation	Ensure that the preservatives on the external surfaces are in good condition and the lid is free of dents and rust stains.
3.	Screen Inspection	Prior to installation of the flanged screen assembly and monthly when in use	Ensure that the screen is undamaged.
4.	ISFSI pad	Annually	Ensure that the ISFSI Pad (raised areas near the VVM) is free of visible cracks or repaired as appropriate, the interface between the ISFSI Pad and the CEC Flange is grouted (or caulked) if necessary, the ISFSI drain system is functional, the ground water collection and removal system (if used) is in working order. Ensure that the subgrade settlement is minimal and unsightly surface cracks in the ISFSI pad have not developed. Implement counter measures to prevent the opening of surface cracks and excessive pad settlement, if observed. The cathodic protection system shall be routinely verified as operable in accordance with the guidance set forth in this FSAR and the CoC.
5.	Shielding Effectiveness Test	As required by the Radiation Protection Program	—

Table 10.4.2 (continued)

MAINTENANCE ACTIVITIES FOR THE HI-STORM UMAX VVM

	Activity	Frequency	Comment
6.	ISFSI Settlement	Every five years	Confirm that the VVM settlement is within the range of the Plant's "best estimate". Implement countermeasures if the settlement is determined to be excessive by the CoC holder.
7.	VVM Air Temperature Monitoring System (if used)	Per Licensee's QA Program and manufacturer's recommendations	—
8.	VVM In-Service Inspection	Annually	Ensure that the vent screen assembly fasteners or weldments remain coated with preservative, the screen is undamaged, all visible external surfaces are free from significant corrosion, and the air passages are not degraded.
10.	Additional VVM In-Service Inspection for Long-Term Interior and Below-grade Degradation: a) Visual inspection of accessible areas for long-term degradation. b) Additional in-service inspection activities include inspection for foreign material accumulation, corrosion (CEC thinning) and insulation degradation	a) Monthly visual inspection of accessible areas. b) Frequencies for additional in-service inspections are determined on a site-specific basis.	Inspection activities shall be commensurate with site-specific conditions. Under site conditions existing at most ISFSI sites, visual inspection of accessible areas is sufficient to determine the general condition of the system.

10.5 CASK IDENTIFICATION

The HI-STORM UMAX VVM and Closure Lids shall be marked with a durable unique identifier to comply with the provisions of 10CFR72.236(k). Each MPC and HI-TRAC transfer cask shall be marked with a model number, identification number (to provide traceability back to documentation), and the empty weight of the item in accordance with the marking requirements specified in 10 CFR 72.236(k).

10.6 REGULATORY COMPLIANCE

The information presented in this chapter fulfills the regulatory requirements pertaining to the testing and maintenance of the HI-STORM UMAX System, resolution of issues concerning adequacy and reliability, and cask identification. This section demonstrates the corresponding compliance information on the HI-STORM UMAX VVM.

a. The program for pre-operational testing and initial operations, as required by 10CFR72.24(p) for the HI-STORM UMAX VVM, is provided in Section 10.3 herein.

b. The maintenance protocol for the HI-STORM UMAX VVM, as specified in §72.236(g), is provided in Section 10.4 herein.

c. The quality assurance requirements on the design, fabrication, and on-site construction of the HI-STORM UMAX VVM commensurate with its ITS designation (as defined in Section 1.5) are invoked through Chapter 14 of this FSAR and summarized in Sections 10.1, 10.2, and 10.3 herein as called for in §72.24(c), §72.122(a), §72.122(f), §72.128(a)(1) and §72.236(l).

d. The provisions of §72.82(d) and §72.162 with respect to acceptance criteria and the appropriate test program to ensure compliance with the acceptance criterion are fulfilled by Section 10.1, et seq., herein.

e. The quality requirements with respect to inspection, testing, and documentation, as set down in §72.212(b)(8), are provided in Section 10.1 herein.

f. The provisions of §72.236(k) with respect to labeling is met as provided in Section 10.1 here-in.

g. The quality requirements with respect to design and testing for methods of criticality control and confinement effectiveness as set down in §72.124(b), §72.236(c), and §72.236(j) are provided in the governing licensing documents applicable to the MPC type to be stored in the UMAX and are not part of the scope of this application.

10.7 REFERENCES

[10.1.1] American Society of Mechanical Engineers, "Boiler and Pressure Vessel Code," Sections II, III, V, IX, and XI, 2010 Edition.

[10.1.2] American Society for Nondestructive Testing, "Personnel Qualification and Certification in Nondestructive Testing," Recommended Practice No. SNT-TC-1A, December 1992.

[10.1.3] American National Standards Institute, Institute for Nuclear Materials Management, "American National Standard for Radioactive Materials - Special Lifting Devices for Shipping Containers Weighing 10,000 Pounds (4500 kilograms) or More", ANSI N14.6, September 1993.

[10.1.4] "NUREG-0612, Control of Heavy Loads at Nuclear Power Plants", US Nuclear Regulatory Commission, 1980.

[10.1.5] American National Standards Institute, Institute for Nuclear Materials Management, "American National Standard for Radioactive Materials Leakage Tests on Packages for Shipment", ANSI N14.5, January 1997.

[10.1.6] Purchase Specification for Metamic-HT, "PS-11", latest revision.

CHAPTER 11: RADIATION PROTECTION

11.0 INTRODUCTION

This chapter contains the design considerations and operational features that are incorporated in the HI-STORM UMAX system to protect plant personnel and the public from exposure to radioactive contamination and ionizing radiation during handling of the loaded MPCs at the ISFSI. Occupational exposure estimate for typical canister handling operations described in Chapter 9 is discussed in this chapter. Chapter 5 presents dose evaluations at 100 m from a single HI-STORM UMAX system. This 100 m dose information from Chapter 5 can be used for ISFSI planning purposes. The information provided in this chapter meets the requirements of NUREG-1536 [11.0.1].

The HI-STORM UMAX is an underground vertical ventilated module (VVM) designed to accommodate all MPC models listed in Table 1.2.1 for storage at an ISFSI. However, this chapter only supports the certification of the MPC-37 and MPC-89. Because of its underground disposition, the radiological dose to plant personnel as well as members of the general public from an operating ISFSI with HI-STORM UMAX VVM assembly is well below those of its aboveground counterpart. Since the determination of off-site doses is necessarily site-specific, similar dose assessment shall be prepared by the licensee as part of implementing the HI-STORM UMAX System in accordance with 10CFR72.212 [11.0.2].

11.1 ENSURING THAT OCCUPATIONAL RADIATION EXPOSURES ARE AS-LOW-AS-REASONABLY-ACHIEVABLE (ALARA)

11.1.1 Policy Considerations

The HI-STORM UMAX has been designed in accordance with 10CFR72 [11.0.2] and maintains radiation exposures ALARA-consistent with 10CFR20 [11.1.1] and the guidance provided in Regulatory Guides 8.8 [11.1.2] and 8.10 [11.1.3]. Licensees using the HI-STORM UMAX system are permitted to utilize and apply their existing site ALARA policies, procedures, and practices for ISFSI activities to ensure that personnel exposure requirements of 10CFR20 [11.1.1] are met. Personnel performing ISFSI operations shall be trained on the operation of the HI-STORM UMAX system, and shall be familiarized with the expected dose rates around the MPC, HI-STORM VVM, and HI-TRAC transfer cask during all phases of loading, storage, and unloading operations. Pre-job ALARA briefings will be held with workers and radiological protection personnel prior to work on or around the HI-STORM UMAX system. Worker dose rate monitoring, in conjunction with trained personnel and well-planned activities, will significantly reduce the overall dose received by the workers. When preparing or making changes to site-specific procedures for ISFSI activities, users shall ensure that ALARA practices are implemented and the 10CFR20 [11.1.1] standards for radiation protection are met in accordance with the site's written commitments.

11.1.2 Radiation Exposure Criteria

The radiological protection criteria that limit exposure to radioactive effluents and direct radiation from an ISFSI using the HI-STORM UMAX system are as follows:

1. 10CFR72.104 [11.0.2] requires that for normal operation and anticipated occurrences, the annual dose equivalent to any real individual located beyond the owner-controlled area boundary must not exceed 25 mrem to the whole body, 75 mrem to the thyroid, and 25 mrem to any other critical organ. This dose would be a result of planned discharges, direct radiation from the ISFSI, and any other radiation from uranium fuel cycle operations in the area. The licensee is responsible for demonstrating site-specific compliance with these requirements. As discussed below, the design features of the HI-STORM UMAX system components are configured to meeting this and other criteria cited below without undue burden to the user.

2. 10CFR72.106 [11.0.2] requires that any individual located on or beyond the nearest owner-controlled area boundary may not receive from any design basis accident the more limiting of a total effective dose equivalent of 5 rem, or the sum of the deep dose equivalent and the committed dose equivalent to any individual organ or tissue (other than the lens of the eye) of 50 rem. The lens dose equivalent shall not exceed 15 rem and the shallow dose equivalent to skin or to any extremity shall not exceed 50 rem. The licensee is responsible for demonstrating site-specific compliance with this requirement.

3. 10CFR20 [11.1.1], Subparts C and D, limit occupational exposure and exposure to individual members of the public. The licensee is responsible for demonstrating site-specific compliance with this requirement.

4. Regulatory Position 2 of Regulatory Guide 8.8 [11.1.2] provides guidance regarding facility and equipment design features. This guidance has been followed in the design of the HI-STORM UMAX storage system as described below:

    - Regulatory Position 2a, regarding access control, is met by locating the ISFSI in a Protected Area in accordance with 10CFR72.212(b)(5)(ii) [11.0.2] Depending on the site-specific ISFSI design, other equivalent measures may be used. Unauthorized access is prevented once a HI-STORM UMAX VVM is loaded in an ISFSI. Due to the passive nature of the system, only limited monitoring is required, thus reducing occupational exposure and supporting ALARA considerations. The licensee is responsible for site-specific compliance with these criteria.

    - Regulatory Position 2b, regarding radiation shielding, is met by the storage cask and transfer cask biological shielding that minimizes personnel exposure, as described in Chapter 5 and in this chapter. Fundamental design considerations that most directly influence occupational exposures with dry storage systems in general and which have been incorporated into the HI-STORM UMAX system design include:

        – system designs that reduce or minimize the number of handling and transfer operations for each spent fuel assembly;

        – system designs that reduce or minimize the number of handling and transfer operations for each MPC loading;

- system designs that maximize fuel capacity, thereby taking advantage of the self-shielding characteristics of the fuel and the reduction in the number of MPCs that must be loaded and handled;

- system designs that minimize planned maintenance requirements;

- system designs that minimize decontamination requirements at ISFSI decommissioning;

- system designs that optimize the placement of shielding with respect to anticipated worker locations and fuel placement;

- thick ISFSI pad and Self-hardening Engineered Subgrade (SES) provide gamma and neutron shielding;

- streaming paths in the HI-STORM UMAX VVM minimized and limited to the air vent passages; and

- low-maintenance design to reduce occupational dose during long-term storage.

- Regulatory Position 2c, regarding process instrumentation and controls, is met since there are no radioactive systems at an ISFSI.
- Regulatory Position 2d, regarding control of airborne contaminants, is met since the HI-STORM UMAX storage system is designed to withstand all design basis conditions to protect the MPC from losing its confinement integrity. As a result, it is reasonable to postulate that no gaseous releases are anticipated. No surface contamination is expected since the exterior of the MPC is delivered in a clean condition when the transfer cask arrives at the ISFSI.
- Regulatory Position 2e, regarding crud control, is not applicable to a HI-STORM UMAX system ISFSI since there are no radioactive systems at an ISFSI that could contain crud.
- Regulatory Position 2f, regarding decontamination, is met since the exterior of the loaded transfer cask is decontaminated prior to being removed from the plant's fuel building.
- Regulatory Position 2g, regarding monitoring of airborne radioactivity, is met since the MPC provides confinement for all design basis conditions. There is no need for monitoring since no airborne radioactivity is anticipated to be released from the casks at an ISFSI.

- Regulatory Position 2h, regarding resin treatment systems, is not applicable to an ISFSI since there are no treatment systems containing radioactive resins at the ISFSI.
- Regulatory Position 2i, regarding other miscellaneous ALARA items, is met since stainless steel is used in the MPC Enclosure Vessel. This material is resistant to the damaging effects of radiation and is well proven in the used fuel cask service. Use of this material quantitatively reduces or eliminates the need to perform maintenance (or replacement) on the primary confinement system.

11.1.3 Operational Considerations

Operational considerations that most directly influence occupational exposures with dry storage systems in general and that have been incorporated into the design of the HI-STORM UMAX system include:

- totally-passive design requiring minimal maintenance and monitoring (other than security monitoring) during storage;
- remotely operated lift yoke and mating device to reduce time operators spend in the vicinity of the loaded MPC;
- descriptive operating procedures that provide guidance to minimize dose and alert workers to possible changing radiological conditions;
- preparation and inspection of the HI-STORM UMAX VVM and the HI-TRAC transfer cask in low-dose areas;
- HI-STORM UMAX VVM temperature monitoring equipment allows remote monitoring of the vent operability surveillance;
- a sequence of short-term operations based on ALARA considerations; and
- use of mock-ups and dry run training to prepare personnel for actual work situations

11.1.4 Auxiliary/Temporary Shielding

In addition to the design and operational features built into the HI-STORM UMAX system components, a number of ancillary shielding devices can be deployed to mitigate occupational dose. Ancillaries are developed on a site-specific basis that further reduce radiation at key work locations and/or allow for a rapid execution of operations to reduce the time personnel spend in the radiation field. Licensees are encouraged to use such ALARA-friendly ancillaries and practices.

To further reduce the occupational and site boundary dose rates, an optional divider shell shield ring (attached to the divider shell) can be implemented. The drawings provided in

Chapter 1 give more information about this shield ring specification. The results presented in this chapter are for the cases without this shield ring. However, in site specific dose evaluations, this shield ring may be credited if it is used.

11.2 RADIATION PROTECTION FEATURES IN THE SYSTEM DESIGN

As shown in Chapter 5, the HI-STORM UMAX system has excellent radiation blockage characteristics. The long-term shielding effectiveness in the HI-STORM UMAX system is assured to an extremely high level of confidence by virtue of its physical configuration, choice of materials, and design embodiment, as summarized below:

a. Absence of penetrations in the VVM body: The CEC has no penetrations and thus has no path that can serve as a conduit for radiation streaming. All penetrations are in the lid and are configured to maximize scattering of photons and neutrons.

b. Axisymmetric penetrations in the lid: As shown in the drawings, the only penetration in the lid – the exit vent – is axisymmetric that precludes a direct "line-of-sight" from the fuel to the outside. Because the air passage in the lid is formed by welded steel shapes, they will remain invariant over time, making their shielding performance reliably constant over years of use.

c. Aging of foundation, subgrade: Even though a very stiff support foundation is specified, some settlement of the foundation is expected. However, any settlement of the foundation would have no deleterious effect on the extent of shielding to the environment.

   Furthermore, because the subgrade is unloaded, except when a transporter is passing over it, the settlement of the ISFSI pad is expected to be minimal over the ISFSI's service life. Any subgrade settlement, however, will result in a corresponding compaction of its material, which will improve the subgrade's shielding capability. The settlement of the subgrade will not result in any new loading on the CEC structure (which is autonomously supported on the foundation) or the Closure Lid, which is autonomously supported on the CEC structure. As required by the system's maintenance program, any visible gap or crevice between the Container Flange and the surface pad shall be filled with grout or caulking for both aesthetic purposes and enhancement of degradation and corrosion mitigation.

d. Effect of Corrosion: It can be readily deduced from the VVM's design that the only surfaces that are not accessible for corrosion monitoring are the bottom face and outer cylindrical surface of the CEC. As discussed in Chapter 8, corrosion mitigation measures for these surfaces are prescribed and expected to provide adequate corrosion mitigation beyond the Design Life of the VVM. Additionally, any CEC or bottom plate corrosion would have a negligible effect on the off-site dose as the majority of the off-site dose contribution is from the inlets, the inlet plenum region and the outlet.

Thus, the potential for reduction in the shielding effectiveness of the HI-STORM UMAX system due to corrosion is not a concern.

e. Water Intrusion: The HI-STORM UMAX system has three barriers against water intrusion in the fuel space, each of which is engineered to independently prevent incursion of water:

i. The Support Foundation pad and the Enclosure shell (see the Drawing package in Section 1.5) provide a robust barrier against seepage of groundwater in the subgrade surrounding the CECs.

ii. The CEC is a thick-walled welded container that has no penetrations in its body through which water may leak in the MPC storage cavities.

iii. The MPC is a stainless steel weldment designed with fully volumetrically examined Section III class 1 butt welds.

f. Materials of Construction: The materials of construction in the underground portion of the CEC are carbon steel and stainless steel, both of which are proven materials of long-term shielding endurance under neutron and gamma fluence levels that are orders of magnitude greater than those present in the CEC. The Closure Lid is comprised of carbon or stainless steel that encases plain concrete. Concrete, like steel, is a proven durable material in a radiation environment that suffers negligible change in its shielding capability over long periods of use in a radiation environment. Therefore, material-degradation-induced reduction in the shielding effectiveness of the VVM is not a credible concern.

In summary, The design features of the HI-STORM UMAX components ensure that the occupational dose as well as off-site dose from the ISFSI will be ALARA.

11.3 ESTIMATED ON-SITE CUMULATIVE DOSE ASSESSMENT

11.3.1 Estimated Exposures for Loading and Unloading Operations

This section discusses the cumulative exposure to personnel performing loading, unloading, and transfer operations using the HI-STORM UMAX system. Additionally, this section provides measured operational dose values from the actual loading campaigns of an above ground system. As the operational aspects of both underground and aboveground systems are quite similar, as discussed below, the realistic campaign dose of the aboveground system would also be applicable to the HI-STORM UMAX.

The operations associated with the use of the HI-STORM UMAX, described in Chapter 9, are quite similar to the operations for all other variations of the HI-STORM 100 and HI-STORM FW systems. In both the aboveground and underground overpack, the MPC is transferred between the HI-TRAC and the overpack and in both cases the lid of the overpack is placed atop the overpack once the HI-TRAC is removed from the overpack. The only significant difference between the aboveground and underground overpack is the position of the HI-TRAC relative to ground level. For the aboveground overpack, the bottom of the HI-TRAC is approximately 18 feet above the ground and for the underground overpack, the bottom of the HI-TRAC is essentially at ground level. From an operations perspective, it will be easier to access the mating device and the pool lid bolts when the HI-TRAC is positioned atop the underground overpack rather than the aboveground overpack. In both cases, the same bolting and unbolting operations around the base of the HI-TRAC must be performed. Therefore, the estimated occupational dose for these scenarios is the same. The fact that the body of the HI-TRAC is closer to the ground when the underground overpack is being loaded will not affect the occupational dose rate since it is assumed that the workers not performing a task are positioned far enough away as to receive minimal dose. Once the MPC transfer is complete and the HI-TRAC has been removed, the lid is placed on the overpack. For the underground overpack, this is a relatively simple operation of lifting the lid and placing it in the correct location. Unlike the aboveground overpack, the lid is not bolted to the body of the overpack. However, the outlet vent cover is installed on the overpack lid after the lid is placed upon the HI-STORM UMAX, an installation that requires bolting. Installation of the outlet vent cover places workers over the lid and adds some time to the operation. As the dose rates on the side of the outlet vent (dose location 2 in Chapter 5) and top of the VVM lid (dose location 3 in Chapter 5) are small (see Tables 5.1.1 and 5.1.2), the operational dose for this operation would be very small compared to the entire MPC transfer into the VVM. Nevertheless, it is recommended that the operators do not spend any unnecessary time on top of the lid to ensure/meet the ALARA principle.

As for the above-ground systems, exposure estimates for loading operations are expected to bound those for unloading operations for the HI-STORM UMAX. This assessment is based on the similarity of many loadings versus operations with the elimination of several of the more dose-intensive operations (such as weld inspections and leakage testing).

In summary, the estimates of occupational exposure for the entire MPC transfer operations from Reference [11.3.2] are directly applicable to the HI-STORM UMAX, and no further evaluations are performed here.

Additionally, experience has shown that the occupational doses are in general significantly lower than those estimated in [11.3.1] and [11.3.2]. To highlight this, typical total occupational doses for 5 cask loadings are listed in Table 11.3.1, while Table 11.3.2 presents the dose of each operational step for Casks 4 and 5 from Table 11.3.1. These are for aboveground systems using a 100 t HI-TRAC. As discussed above, since the operational sequence for loading the HI-STORM UMAX is essentially the same as for the above ground systems, similar values would be expected for the HI-STORM UMAX.

11.3.2 Excavation Activities

In the event it is desired to expand an ISFSI utilizing the HI-STORM UMAX design, excavation of material (i.e., soil) is required. Radiation protection of the excavation activities is achieved by prescribing a minimum proximity of any excavation to an existing HI-STORM UMAX array. Site specific radiation protection measures for excavation activities need to include confirmation of the minimum SES properties along with the minimum distances between the excavation area and the loaded VVMs, as well as radiological monitoring of the excavation area.

Site specific evaluations also need to be performed to ensure that the radiation protection space boundary is maintained. Site specific accident scenarios (e.g., seismic conditions) will need to be accounted for in these evaluations. Table 5.4.4 presents a representative dose rate at the surface of the radiation protection space. Additionally, a general accident scenario evaluation has been performed for the HI-STORM UMAX design. The impact of a tornado missile penetrating the SES creating a horizontal hole extending 5.5 feet from the external surface of the radiation protection boundary was considered. This evaluation, presented in Table 5.1.4, demonstrates that the dose at the site boundary is below the limit specified in 10 CFR 72.

11.3.3 Normal Operation of Storage

During normal operation of storage, radiation will predominantly emanate from the inlet and outlet vents and the top of the lid. However, there are also some additional radiations streaming paths and scenarios that may have to be considered in the radiation protection program. The following two scenarios have been evaluated for the HI-STORM UMAX design.

The first scenario evaluated address radiation streaming from a loaded VVM through an adjacent empty VVM. An empty VVM adjacent to a loaded VVM could potentially constitute a radiation streaming path since the SES providing shielding is limited between adjacent VVMs. Therefore, radiation passing through the SES to the unloaded VVM will have a path of less shielding and could contribute to occupational dose. This evaluation is presented in Chapter 5, and concluded that there are no concerns about the dose rates

contributing to occupational dose across the top of the empty VVM due to radiation streaming from the loaded neighboring VVM.

The second scenario concerns the SES access tube, or test station, that is part of the ICCPS design and could represent a potential streaming path. Therefore, radiation passing through the SES access tube could contribute to occupational dose. This evaluation is presented in Chapter 5, and assumes a tube located about 5.5 feet from the center of the VVM with a diameter of 4 inches, that reaches down to the support foundation. With these dimensions, it is shown that there are no concerns about the dose rates contributing to occupational dose on the top of the SES access tube due to radiation streaming from a loaded VVM. However, if the tube is larger or located closer to the VVM, then the actual dimensions should be considered in the site specific dose rate calculations, and the result of the calculations should be considered in the site specific radiation protection program.

11.3.4 Estimated Exposures for Surveillance and Maintenance

Because of its low profile, the surveillance at a HI-STORM UMAX ISFSI can be performed without physically walking between the VVMs and therefore, occupational exposure required for security surveillance and maintenance will be bounded by above ground systems [11.3.2]. Typical estimates of the occupational exposure required for security surveillance and maintenance of an ISFSI can be found in References [11.3.1][11.3.2]. Security surveillance time is based on a daily security patrol around the perimeter of the ISFSI security fence. Users may opt to utilize electronic temperature monitoring of the HI-STORM UMAX modules or remote viewing methods instead of performing direct visual observation of the modules. Although the HI-STORM UMAX system requires only minimal maintenance during storage (e.g., touch-up paint), maintenance will be required around the ISFSI for items such as security equipment maintenance, grass cutting, snow removal, vent system surveillance, drainage system maintenance, and lighting, telephone, and intercom repair. Such infrequent activities are not included in the site boundary dose assessment.

Table 11.3.1

MEASURED TOTAL OPERATIONAL DOSE FOR A LOADING CAMPAIGN USING HI-STORM SYSTEMS

Cask Number	Total Heat Load (kW)	Total Campaign Dose (mrem)
1	23.79	370
2	24.96	208
3	25.70	183
4	25.75	164
5	26.36	187

Table 11.3.2 MEASURED OPERATIONAL DOSE FOR DIFFERENT TASKS FOR CASKS 4 AND 5 IN TABLE 11.3.1		
Task Description	Cask 4 (mrem)	Cask 5 (mrem)
	25.75 kW	26.36 kW
Preparations and placement of HI-TRAC in spent fuel pool	2	3
Fuel loading and verification	13	8
Placement of HI-TRAC in the decontamination pit and decontamination	35	34
Welding of MPC lid	14	18
Drying operation	20	19
Welding of the closure plates	33	34
Stackup operation	43	60
Move HI-STORM to ISFSI	4	11
Total dose	**164**	**187**

11.4 REFERENCES

[11.0.1] NUREG-1536, “Standard Review Plan for Dry Cask Storage Systems”, U.S. Nuclear Regulatory Commission, January 1997.

[11.0.2] *U.S. Code of Federal Regulations,* Title 10, "Energy" Part 72 "Licensing Requirements for Independent Storage of Spent Nuclear Fuel and High-Level Radioactive Waste".

[11.1.1] *U.S. Code of Federal Regulations,* Title 10, "Energy" Part 20 "Standards for Protection Against Radiation".

[11.1.2] U.S. Nuclear Regulatory Commission "Information Relevant to Ensuring that Occupational Radiation Exposures at Nuclear Power at Nuclear Power Stations will be As Low As Reasonably Achievable", Regulatory Guide 8.8, June 1978.

[11.1.3] U.S. Nuclear Regulatory Commission, "Operating Philosophy for Maintaining Occupational Radiation Exposures As Low As is Reasonably Achievable", Regulatory Guide 8.10, Revision 1-R, May 1997.

[11.3.1] “Final Safety Analysis Report for the HI-STORM 100 Cask System”, HI-2002444,USNRC Docket 72-1014.

[11.3.2] “Final Safety Analysis Report for the HI-STORM FW MPC Storage System”, Holtec Report # 2114830,USNRC Docket #72-1032.

CHAPTER 12: ACCIDENT EVALUATION

12.0 INTRODUCTION

This chapter is focused on the safety evaluation of all off-normal and accident events germane to the HI-STORM UMAX vertical ventilated module (VVM) containing a loaded Multi-purpose Canister (MPC). For each postulated event, the event cause, means of detection, consequences, and corrective actions, as applicable, are discussed and evaluated. For other miscellaneous events (i.e., those not categorized as either design basis off-normal or accident condition events), a similar outline for safety analysis is followed. As applicable, the evaluation of consequences includes the impact on the structural, thermal, shielding, criticality, confinement, and radiation protection performance of the system due to each postulated event.

The analyses summarized in this chapter focus on the governing canisters out of the population of MPCs listed in Table 1.2.1. This chapter, however, supports the certification of only MPC-37 and MPC-89 at this time. The analyses reported for smaller canisters are for reference purposes only.

The structural, thermal, shielding, criticality, and confinement features and performance of the HI-STORM UMAX system under the short-term operations and various conditions of storage are discussed in Chapters 3, 4, 5, 6, and 7. The evaluations provided in this chapter are based on the design features and analyses reported therein. The accidents considered in this chapter follow the guidance in NUREG-1536.

Technical descriptions and safety analyses pertaining to the components common to the "FW" and the "UMAX" systems are referenced in this FSAR, as appropriate, to the HI-STORM FW FSAR. To facilitate convenient access to the referenced material, the latest edition of the HI-STORM FW FSAR has been placed in this docket and a list of "FW" FSAR sections germane to this chapter is provided in Table 12.0.1 herein.

TABLE 12.0.1

APPLICABLE SECTIONS OF HI-STORM FW FSAR*

Location of UMAX FSAR	**Subject of the Reference**	**Location in HI-STORM FW FSAR, Revision 3**
Sub-Section 12.1.1	Structural evaluation of the MPC enclosure vessel for off-normal internal pressure conditions	Sub-Section 3.4.4
Sub-Section 12.1.2	Limitations on the use of the HI-TRAC VW cask for loading MPCs under off-normal thermal conditions	Sub-Section 12.1.2
Sub-Section 12.1.3	Leakage of one MPC seal weld	Sub-Section 12.1.3
Sub-Section 12.1.6	FHD malfunction evaluations	Sub-Section 12.1.5
Sub-Section 12.2.1	Fire accident evaluation	Sub-Section 12.2.4
Sub-Section 12.2.2	Partial blockage of MPC basket vent holes	Sub-Section 12.2.5
Sub-section 12.2.3	Tornado analysis for HI-TRAC VW	Sub-Section 12.2.6
Sub-Section 12.2.7	Confinement Boundary Leakage	Section 7.1
Sub-Section 12.2.13	HI-TRAC VW handling accidents	Sub-Section 12.2.1
Sub-Section 12.3.2	MPC reflood evaluation	Sub-Section 12.3.1

* For convenience of reference, the specific revision of the HI-STORM FW FSAR that is referenced in the safety analysis herein is placed in this docket. Updated versions of the HI-STORM FW FSAR shall be placed in this docket as necessary so as to ensure that the safety analyses on the "UMAX" docket (72-1040) remain aligned with the material referenced in the HI-STORM FW FSAR.

12.1 OFF-NORMAL CONDITIONS

Off-normal conditions, as defined in accordance with ANSI/ANS-57.9, are those conditions which, although not occurring regularly, are expected to occur no more than once a year. In this section, design events pertaining to off-normal operation for expected operational occurrences are considered. The off-normal conditions are described in Section 2.3.

The following off-normal events are applicable to the HI-STORM UMAX system:

- Off-Normal Pressure
- Off-Normal Environmental Temperature
- Leakage of One Seal
- Partial Blockage of the Air Inlet Plenum
- Hypothetical Non-Quiescent Wind
- FHD Malfunction

The results of the evaluations presented herein demonstrate that the HI-STORM UMAX System can withstand the effects of off-normal events and remain in compliance with the applicable acceptance criteria.

12.1.1 Off-Normal Pressure

The sole pressure boundary in the HI-STORM UMAX storage System is the MPC enclosure vessel. The off-normal pressure condition is specified in Section 2.3. The off-normal pressure for the MPC internal cavity is a function of the initial helium fill pressure and the steady state temperature reached within the MPC cavity under normal ambient temperature. The MPC internal pressure under the off-normal condition is evaluated with 10% of the fuel rods ruptured and with 100% of ruptured rods fill gas and 30% of ruptured rods fission gases released to the cavity.

12.1.1.1 Postulated Cause of Off-Normal Pressure

After fuel assembly loading, the MPC is drained, dried, and backfilled with an inert gas (helium) to assure long-term fuel cladding integrity during dry storage. Therefore, the probability of failure of intact fuel rods in dry storage is extremely low. Nonetheless, the event is postulated and evaluated.

12.1.1.2 Detection of Off-Normal Pressure

The HI-STORM UMAX system is designed to withstand the MPC off-normal internal pressure without any effects on its ability to meet its safety requirements. There is no requirement or safety imperative for detection of off-normal pressure and, therefore, no monitoring is required.

12.1.1.3 Analysis of Effects and Consequences of Off-Normal Pressure

The MPC off-normal internal pressure is reported in Section 4.6.1.4 for the limiting fuel storage scenario wherein the canister pressurized to the technical specification maximum helium backfill pressure sustains a 10% rod rupture that causes a 100% of the ruptured rod fill gas and 30% of the ruptured rod gaseous fission products released into the MPC cavity.

The analysis of the above scenario shows that the MPC pressure remains below the design MPC internal pressure (given in Table 2.3.5).

i. Structural

The structural evaluation of the MPC enclosure vessel for off-normal internal pressure conditions is discussed in the HI-STORM FW FSAR [4.1.2]. The stresses resulting from the off-normal pressure are confirmed to be bounded by the applicable pressure boundary stress limits.

ii. Thermal

The MPC internal pressure for off-normal conditions is reported in Table 4.4.7. The design basis internal pressure used in the structural evaluation (Table 2.3.5) bounds the computed off-normal condition pressure.

iii. Shielding

There is no effect on the shielding performance of the system as a result of this off-normal event.

iv. Criticality

There is no effect on the criticality control features of the system as a result of this off-normal event.

v. Confinement

There is no effect on the confinement function of the MPC as a result of this off-normal event. As discussed in the structural evaluation above, all pressure boundary stresses remain within allowable ASME Code values, assuring Confinement Boundary integrity.

vi. Radiation Protection

Since there is no degradation in shielding or confinement capabilities as discussed above, there is no effect on occupational or public exposures as a result of this off-normal event.

12.1.1.4 Corrective Action for Off-Normal Pressure

The HI-STORM UMAX system is designed to withstand the off-normal pressure without any effects on its ability to maintain safe storage conditions. Therefore, there is no corrective action requirement for off-normal pressure.

12.1.1.5 Radiological Impact of Off-Normal Pressure

The event of off-normal pressure has no radiological impact because the confinement barrier and shielding integrity are not affected.

12.1.1.6 Conclusion

Based on this evaluation, it is concluded that the off-normal pressure does not affect the safe operation of the HI-STORM UMAX system.

12.1.2 Off-Normal Environmental Temperatures

The HI-STORM UMAX System is designed for use at any site in the United States. Off-normal environmental temperatures have been conservatively selected to bound the environmental temperatures at all candidate sites in the United States (See Subsection 2.3.3 for definition of the term off-normal environmental temperature). The off-normal temperature limits are reported in Table 2.3.6.

12.1.2.1 Postulated Cause of Off-Normal Environmental Temperatures

The off-normal environmental temperature is postulated as a constant ambient temperature caused by extreme weather conditions. To determine the effects of the off-normal temperatures, it is conservatively assumed that these temperatures persist for a sufficient duration to allow the HI-STORM UMAX System to achieve thermal equilibrium. Because of the large mass of the HI-STORM UMAX System with its corresponding large thermal inertia and the limited duration for the off-normal temperatures, this assumption is conservative.

12.1.2.2 Detection of Off-Normal Environmental Temperatures

The HI-STORM UMAX System is designed to withstand the off-normal environmental temperatures without any effects on its ability to maintain safe storage conditions. There is no requirement for detection of off-normal environmental temperatures for the HI-STORM UMAX overpack and MPC. The limitations on the use of the transfer cask for loading canisters in the HI-STORM UMAX system under off-normal thermal conditions are contained in Section 12.1.2 of the HI-STORM FW FSAR which must be observed.

12.1.2.3 Analysis of Effects and Consequences of Off-Normal Environmental Temperatures

The off-normal event is considered to be characterized by an off-normal environmental temperature with insolation for sufficient duration to reach thermal equilibrium. The evaluation is performed for a limiting fuel storage configuration. The off-Normal ambient temperature condition is evaluated in Subsection 4.6.1.1. The results are in compliance with off-normal pressure and temperature limits in Tables 2.3.5 and 2.3.7 in Chapter 2.

The off-normal event considering an environmental temperature of -40°F and no solar insolation for a sufficient duration to reach thermal equilibrium is evaluated with respect to material design temperatures of the HI-STORM UMAX VVM. The HI-STORM UMAX VVM structure is conservatively assumed to reach the extreme cold condition (-40°F) throughout its body. The qualification of the VVM structure under the extreme cold condition is provided in Chapter 8.

i. Structural

The effect on the MPC for the upper off-normal thermal conditions is an increase in the internal pressure. As shown in Section 4.6.1.1, the resultant pressure is below the off-normal design pressure (Table 2.3.5). The stresses resulting from the off-normal pressure are confirmed to be bounded by the applicable pressure boundary stress limits. The effect of the lower off-normal thermal conditions (i.e., -40°F) requires an evaluation of the potential for brittle fracture. Such an evaluation is presented in Chapter 8.

ii. Thermal

The resulting off-normal system and fuel assembly cladding temperatures for the hot conditions are provided in Table 4.6.1. This evaluation indicates that all temperatures for the off-normal environmental temperatures event are within the allowable values for off-normal conditions listed in Table 2.3.7. Additionally, the increased temperatures generate an elevated MPC internal pressure, reported in Table 4.6.5, which is less than the off-normal design pressure limit specified in Table 2.3.5. The temperatures and pressures resulting from the off-normal ambient temperature event are confirmed to be bounded by

the applicable system temperature and pressure limits; therefore, there is no adverse effect on the system's thermal function.

iii. Shielding

There is no effect on the shielding performance of the system as a result of this off-normal event.

iv. Criticality

There is no effect on the criticality control features of the system as a result of this off-normal event.

v. Confinement

There is no effect on the confinement function of the MPC as a result of this off-normal event. As discussed in the structural evaluation above, all pressure boundary stresses in the canister remain within allowable ASME Code values, assuring Confinement Boundary integrity.

vi. Radiation Protection

Since there is no degradation in shielding or confinement capabilities as discussed above, there is no effect on occupational or public exposures as a result of this off-normal event.

12.1.2.4 Corrective Action for Off-Normal Environmental Temperatures

The HI-STORM UMAX System is designed to withstand the off-normal environmental temperatures without any effects on its ability to maintain safe storage conditions. As required by the HI-STORM FW FSAR [4.1.2], for ambient temperatures from 0° to 32°F, ethylene glycol fortified water must be used in the water jacket of the HI-TRAC VW transfer cask to prevent freezing. There are no corrective actions required for off-normal environmental temperatures.

12.1.2.5 Radiological Impact of Off-Normal Environmental Temperatures

Off-normal environmental temperatures have no radiological impact, as the confinement barrier and shielding integrity are not affected.

12.1.2.6 Conclusion

Based on the above evaluation, it is concluded that the specified off-normal environmental temperatures do not affect the safe operation of the HI-STORM UMAX System.

12.1.3 Leakage of One MPC Seal Weld

Leakage of one MPC seal weld evaluated in the HI-STORM FW FSAR Section 12.1.3 [4.1.2] is incorporated by reference.

12.1.4 Partial Blockage of Air Inlet Plenum

Partial blockage (50%) of the air intake system has been postulated as an off-normal event in Section 2.5.

The HI-STORM UMAX intake ducts are designed with debris screens, as is the outlet vent flue located in the Closure Lid. These screens protect the openings from the incursion of foreign objects. However, as required by the design criteria presented in Chapter 2, it is conservatively assumed that 50% of the air inlet opening is completely blocked. The scenario of the partial blockage of air inlets is evaluated with a normal ambient temperature (Table 2.3.6), full solar insolation, and Design Basis SNF decay heat value case. This condition is analyzed in Chapter 4 to demonstrate the acceptability of the system thermal performance during this event.

12.1.4.1 Postulated Cause of Partial Blockage of Air Inlets

The presence of screens prevents foreign objects from entering the openings and the screens are either inspected periodically or the system temperature field is monitored per the technical specifications. It is, however, possible that blowing debris may partially block the inlet openings for a short time until the openings are cleared of debris.

12.1.4.2 Detection of Partial Blockage of Air Inlet

The detection of the partial blockage of air inlet openings will occur during the routine visual inspection of the screens or temperature monitoring of the outlet air required by the technical specifications. The frequency of inspection is based on an assumed complete blockage of all air inlet openings. There is no inspection requirement as a result of the postulated partial inlet blockage because the complete blockage of all air inlet openings is bounding.

12.1.4.3 Analysis of Effects and Consequences of Partial Blockage of Air Inlets

i. Structural

The effect of partial blockage of the air inlet plenum on the MPC is an increase in component and fuel cladding temperatures and internal pressure and thus an increase in pressure boundary stresses. However, the resultant temperatures and pressures are below the off-normal design limits as discussed in the thermal effects evaluation below. The MPC stresses resulting from the partial blockage of air inlets are confirmed to be bounded by the applicable pressure boundary stress limits; therefore, there is no effect on structural function.

In summary, there are no structural consequences as a result of this off-normal event since the HI-STORM UMAX components do not exceed the off-normal temperature limits (Table 2.3.7).

ii. Thermal

The thermal evaluation of partial blockage of air inlet is discussed in Subsection 4.6.1.2. The calculated bounding temperatures conservatively evaluated as a 50% blockage of sufficient duration to reach the asymptotic maximum (steady-state) temperature field are reported in Table 4.6.1 and below the MPC and VVM off-normal design temperature limits specified in Table 2.3.7, as applicable. Additionally, the increased temperatures generate an elevated MPC internal pressure, reported in Table 4.6.5, which is less than the off-normal design pressure limit specified in Table 2.3.5. The temperatures and pressures resulting from the partial blockage of air inlet event are confirmed to be bounded by the applicable system temperature and pressure limits; therefore, there is no adverse effect on the system's thermal function.

iii. Shielding

There is no adverse effect on the function of shielding features of storage the system as a result of this off-normal event.

iv. Criticality

There is no effect on the criticality control features of the system as a result of this off-normal event.

v. Confinement

There is no effect on the confinement function of the MPC as a result of this off-normal event.

vi. Radiation Protection

Since there is no degradation in shielding or confinement capabilities as discussed above, there is no predicted adverse effect on occupational or public exposures as a result of this off-normal event.

12.1.4.4 Corrective Action for Partial Blockage of Air Inlets

The corrective action for the partial blockage of air inlet openings is the removal, cleaning, and replacement of the affected mesh screens. After clearing of the blockage, the storage module

temperatures will return to the normal temperatures reported in Chapter 4. Partial blockage of air inlet openings does not affect the safe operation of the HI-STORM UMAX System.

Periodic inspection of the HI-STORM UMAX air opening screens is required per the technical specifications. Alternatively, per the technical specifications, the outlet air temperature is monitored. The frequency of inspection is based on an assumed blockage of all air inlet openings analyzed in Section 12.2.

12.1.4.5 Radiological Impact of Partial Blockage of Air Inlets

The off-normal event of partial blockage of the air inlet opening has no radiological impact because the confinement barrier is not breached and the system's shielding effectiveness is not diminished.

12.1.4.6 Conclusion

Based on the above evaluation, it is concluded that the off-normal partial blockage of air inlet ducts event does not affect the safe operation of the HI-STORM UMAX VVM.

12.1.5 Hypothetical Non-Quiescent Wind

12.1.5.1 Cause of Event

Wind is a meteorological event that occurs in every area in the world. The normal condition of storage of the canisters assumes quiescent ambient conditions with an annular average temperature that bounds the historical data for all sites in the United States. The hypothetical non-quiescent wind condition is intended to simulate the thermal response of a site subject to a persistent wind event that tends to disrupt the heat rejection performance of the system.

12.1.5.2 Simulation of the Event

The HI-STORM UMAX storage system is designed to store the canisters listed in Table 1.2.1 at any ISFSI site in the United States in compliance with this FSAR. Chapter 4 evaluates the effects of low speed wind postulated as a constant horizontal wind caused by hypothetical weather conditions (see Section 4.6.1.3). To determine the effects of this hypothetical wind event, it is conservatively assumed that the wind persists in a fixed direction for a sufficient duration to allow the HI-STORM UMAX system to reach thermal equilibrium. Temperature results are compared with off-normal temperature limits. Because of the large mass of the HI-STORM UMAX System with its corresponding large thermal inertia and the unlikely condition of a unidirectional wind existing for a long period of time, this assumption is extremely conservative.

12.1.5.3 Safety Analysis

The following is an evaluation of effects on structural, thermal, criticality, confinement, and radiation protection performance on the HI-STORM UMAX storage system:

i. Structural

The effect on the MPC for the hypothetical non-quiescent wind condition is an increase in component and fuel cladding temperatures and internal pressure and thus an increase in pressure boundary stresses. However, the resultant temperatures and pressures are below the off-normal design limits as discussed in the thermal effects evaluation below. The MPC stresses resulting from the off-normal temperature event are confirmed to be bounded by the applicable pressure boundary stress limits; therefore, there is no effect on structural function.

ii. Thermal

Chapter 4 calculates peak fuel cladding temperatures as a function of the horizontal wind speed. The calculated maximum peak cladding temperatures reported in Table 4.6.4 are below the off-normal limits specified in Table 2.3.7 for both moderate burnup and high burnup fuel. Additionally, the elevated MPC internal pressure is reported in Table 4.6.5, which is less than the off-normal design pressure limit specified in Table 2.3.5. By this evaluation, temperatures and pressures resulting from the hypothetical non-quiescent wind event are confirmed to be bounded by the applicable system off-normal temperature and pressure limits; therefore, there is no effect on thermal function.

iii. Shielding

There is no effect on the function of shielding features of the system as a result of this event.

iv. Criticality

There is no effect on the function criticality control features of the MPC as a result of this event.

v. Confinement

There is no effect on the function of confinement features of the MPC as a result of this event.

vi. Radiation Protection and Consequences

Since there is no effect on shielding or confinement functions as discussed above, there is no radiological consequence (from effluents and direct radiation) and no expected increase to occupational exposures as a result of this event.

12.1.5.4 Corrective Action and Counter-measures

Because the HI-STORM UMAX System is designed to withstand the hypothetical non-quiescent wind event without any effect on its ability to maintain safe storage conditions, there is no requirement for detection and counter-measures.

12.1.5.5 Conclusion

Based on this evaluation, it is concluded that the hypothetical non-quiescent wind event does not affect the safe operation of the HI-STORM UMAX VVMs.

12.1.6 FHD Malfunction

FHD malfunction evaluated in the HI-STORM FW FSAR Section 12.1.5 [4.1.2] is incorporated by reference.

12.2 ACCIDENT EVENTS

Accidents, in accordance with ANSI/ANS-57.9, are either infrequent events that could reasonably be expected to occur during the lifetime of the HI-STORM UMAX system or events postulated because their consequences may affect the public health and safety. Sections 2.4 and 2.5, respectively, define the structurally and thermally significant loadings that are classified design basis accidents. These design basis accident events have been evaluated in this FSAR to quantify the safety margins in the storage system.

The load combinations evaluated for postulated accident conditions are defined in Chapter 2. The structural qualification of accidents is provided in Chapter 3.

The following accident events germane to the safety evaluation of HI-STORM UMAX system are identified by reference to Sections 2.4 and 2.5:

- Fire Accident
- Partial Blockage of MPC Basket Vent Holes in long- term storage
- Tornado
- Flood
- Earthquake
- 100% Fuel Rod Rupture
- Confinement Boundary Leakage
- Explosion
- Lightning
- 100% Blockage of Air Inlets
- Burial Under Debris
- Extreme Environmental Temperature
- HI-TRAC VW Transfer Cask Handling Accident

The results of the evaluations performed in this FSAR demonstrate that the HI-STORM UMAX storage system can withstand the effects of all credible and hypothetical accident conditions and natural phenomena without affecting its safety function, In the following, the evaluation of the design basis postulated accident conditions and natural phenomena is presented which demonstrates that the requirements of 10CFR72.122 and of 10 CFR72.106(b) and 10CFR20 are met.

12.2.1 Design Basis Fire Event (Load Case 5 in Section 2.4)

12.2.1.1 Cause of Fire

The potential of a fire accident near an ISFSI pad is considered to be rendered extremely remote by ensuring that there are no significant combustible materials in the area. The only credible concern is related to a transport vehicle fuel tank fire engulfing a loaded HI-STORM UMAX

VVM or a HI-TRAC VW transfer cask. HI-TRAC VW transfer cask fire evaluated in the HI-STORM FW FSAR Section 12.2.4 [4.1.2] is incorporated by reference. HI-STORM UMAX fire is evaluated in the following.

12.2.1.2 Fire Analysis

The HI-STORM UMAX System must withstand elevated temperatures under the Design Basis Fire event defined in Table 2.3.1. The acceptance criteria for the fire accident are provided in Section 2.3 and the thermal analysis is contained in Section 4.6.2.1.

i. Structural

The effect of the fire accident on the HI-STORM UMAX system is an increase in fuel cladding and system component temperatures and MPC internal pressure and thus an increase in MPC pressure boundary stresses. However, the resultant temperatures and pressures are below the accident design limits as discussed below. The MPC stresses resulting from the fire accident event are confirmed to be bounded by the applicable pressure boundary stress limits; therefore, there is no effect on structural function.

ii. Thermal

As discussed in Chapter 4, the effect of the fire does not cause any system component or the contained fuel to exceed any limit set in this FSAR. The Design Basis Fire has a negligible impact on MPC pressure. The temperatures and pressures resulting from the fire accident event are confirmed to be bounded by the applicable system temperature and pressure limits; therefore, there is no deleterious effect on the system's thermal function.

iii. Shielding

The loss of shielding, if any, has been determined by scoping calculations to be of insignificant consequence in Chapter 4. With respect to concrete damage from a fire, NUREG-1536 (4.0,V,5.b) states: "the loss of a small amount of shielding material is not expected to cause a storage system to exceed the regulatory requirements in 10 CFR 72.106 and, therefore, need not be estimated or evaluated in the FSAR.

Less than 5% of the Closure Lid concrete thickness is computed to exceed the short-term temperature limit therefore the effect of this small amount of degraded (not lost) shielding material is not estimated or evaluated in this FSAR.

iv Criticality

There is no effect on the criticality control features of the system as a result of this event.

v. Confinement

There is no effect on the confinement function of the MPC as a result of this event since the structural integrity of the confinement boundary is unaffected.

vi. Radiation Protection

Since there is minimal reduction, if any, in shielding and no effect on the confinement capabilities as discussed above, there is no effect on occupational or public exposures as a result of this accident event.

12.2.1.3 Fire Accident Corrective Actions

Upon detection of a fire adjacent to a loaded HI-STORM UMAX VVM, the ISFSI owner shall take the appropriate immediate actions necessary to extinguish the fire. Following the termination of the fire, a visual and radiological inspection of the equipment shall be performed.

If damage to the HI-STORM UMAX VVM as the result of a fire event is widespread, and/or as radiological conditions require (based on dose rate measurements), the MPC shall be removed from the HI-STORM UMAX VVM in accordance with the procedure set down in Chapter 9. However, the thermal analysis described herein demonstrates that only a limited amount of lid concrete which is behind the steel enclosure exceeds its design temperature. The HI-STORM UMAX VVM may be returned to service after appropriate restoration (reapplication of coatings, etc.) if there is no significant increase in the measured dose rates (i.e., the shielding effectiveness of the overpack is confirmed) and if the visual inspection is satisfactory.

There is no effect on the function of criticality control features of the MPC as a result of this accident event.

Based on the foregoing evaluation, it is concluded that the overpack fire accident does not affect the safe operation of the HI-STORM UMAX VVMs.

12.2.1.4 Conclusion

Based on the above evaluation, it is concluded that the Design Basis Fire accident does not affect the safe operation of the HI-STORM UMAX System.

12.2.2 Partial Blockage of MPC Basket Vent Holes

Partial blockage of MPC basket vent holes evaluated in the HI-STORM FW FSAR Section 12.2.5 [4.1.2] is incorporated by reference.

12.2.3 Tornado (Load Case 02 in Section 2.4)

12.2.3.1 Causal Factors

Tornado and high winds are principally caused by the uneven heating of the earth's atmosphere, coupled with gravitational forces and the rotation of the earth. The HI-STORM UMAX System will be deployed in an open area environment and thus will be subject to ambient environmental conditions throughout the storage period. Additionally, the transfer of the MPC between the HI-TRAC VW transfer cask and the storage overpack may be performed at the unsheltered ISFSI concrete pad. It is therefore possible that the HI-STORM UMAX storage system may experience the extreme environmental conditions resulting in the impact from a tornado-borne projectile.

12.2.3.2 Tornado Analysis

A tornado event is characterized by high wind velocities and tornado-generated missiles. The reference missiles considered in this FSAR (see Table 2.3.3) are of three sizes: small, medium, and large. A small projectile, upon collision with a cask, would tend to penetrate it. A large projectile, such as an automobile, on the other hand, would tend to cause deformation.

Because of its underground construction, the HI-STORM UMAX is not subject to overturning action by the tornado wind. The effect of tornado missiles propelled by high velocity winds that attempt to penetrate the exposed portions of the HI-STORM UMAX must, however, be considered.

The tornado analysis for a HI-TRAC VW transfer cask evaluated in the HI-STORM FW FSAR Section 12.2.6 [4.1.2] is incorporated by reference.

The evaluation of effects on structural, thermal, criticality, confinement, and radiation protection performance on the HI-STORM UMAX system is summarized below.

i. Structural

Analyses presented in Chapter 3 show that the impact of large and intermediate tornado missiles (see Table 2.3.3) on the HI-STORM UMAX closure lid does not result in the perforation of the lid or result in a structural collapse of the lid. The sole effect of the tornado missile impact on the HI-STORM UMAX VVM is some minor global deformation of the VVM Closure Lid under the large missile and some localized deformation of the VVM Closure Lid under the intermediate missile. All Design Basis missiles are found to be stopped by the VVM assembly before reaching the MPC stored inside. Therefore, MPC damage by impact from a Design Basis Missile is ruled out.

ii. Thermal

There is no effect on the function of HI-STORM UMAX VVM heat transfer features as a result of this accident event. The deformations in the VVM Closure Lid due to missile impact are minor relative to the available area of the flow opening.

iii. Shielding

Certain tornado missile scenarios may result in shielding degradation of the HI-STORM UMAX Closure Lid; however, the overall shielding effect will be negligible due to the sheer size and mass of the Closure Lid. (The HI-STORM UMAX VVM is heavily shielded (a thick MPC lid protected by the monolithic steel-concrete-steel VVM Closure Lid weighing in excess of 30,000 lbs; see Section 3.2.)

iv. Criticality

There is no effect on the criticality control features of the MPC as a result of this accident event.

v. Confinement

There is no effect on the confinement function of the MPC as a result of this accident event.

12.2.3.3 Radiation Protection and Consequences

There is no effect on shielding or confinement functions of the MPC. The effect on shielding function of the VVM Closure Lid is negligible and therefore the radiological impact on the site boundary dose will be negligible. Thus, a negligible increase in radiological consequence (from effluents and direct radiation) is expected as a result of this accident. A minor increase to occupational exposures for the performance of corrective actions is also expected.

12.2.3.4 Tornado Accident Corrective Action

Following exposure of the HI-STORM UMAX System to a tornado, the ISFSI owner shall perform a visual and radiological inspection of the facility.

Damage sustained by the VVMs or vent screens shall be inspected and may be repaired, if required, while in-service. The system may continue in service after appropriate restoration (reapplication of coatings, etc.) if there is no significant increase in the measured dose rates (i.e., the system's shielding effectiveness is confirmed) and if the final visual inspection is satisfactory.

12.2.3.5 Conclusion

Based on the above evaluation, it is concluded that the Design Basis tornado accident will not affect the safe operation of the HI-STORM UMAX System.

12.2.4 Flood (Load Case 7 in Table 2.4.1)

12.2.4.1 Cause of Flood

Many ISFSIs are located in flood plains susceptible to floods. Therefore, it is necessary for such ISFSIs to define a Design Basis Flood (DBF). The potential sources for the floodwater may be swelling rivers or streams from heavy rains or rapid melting of upstream snow, tsunami, dam break, earthquake, hurricane, etc.

12.2.4.2 Analysis

Because of its underground construction, the HI-STORM UMAX is not subject to overturning action by moving floodwater. The permissible height of floodwater for storing an MPC is governed by the design basis flood defined in Table 2.4.1. An MPC not qualified to withstand the external pressure from a site's Design Basis Flood event shall not deployed in the HI-STORM UMAX system.

The following is an evaluation of effects on structural, thermal, criticality, confinement, and radiation protection performance on the HI-STORM UMAX system.

i. Structural

Unlike free standing casks, moving flood water is not an event of safety consequence to the HI-STORM UMAX: The buried configuration of the HI-STORM UMAX system renders it immune from sliding (that is germane to above ground freestanding casks) under the action of a design basis flood. Since the CEC shell is directly in contact with the Self-hardening Engineered Subgrade, there is no risk of a global deformation of the CEC under the DBF event.

ii. Thermal

The flooded HI-STORM UMAX ISFSI will reject heat to the floodwater. Because the heat transfer coefficient in water is considerably greater than that under the ventilation air, the temperature of the contents will be lowered. Furthermore, the heat dissipated from MPC and MPC pedestal flood tends to boil the flood water entering "UMAX" cavity and lower water level to restore sufficient air ventilation flow. Thus, the thermal effect of flood is actually salutary for the system's performance.

Partial blockage of the bottom cutout is evaluated in Section 4.6.2.5. The resulting maximum temperatures are provided in Table 4.6.9, and confirmed to maintain below the accident temperature limits specified in Table 2.3.7. In the case when flood water/soil is just high enough to completely block the divider shell cutout, it is bounded by the 100% inlet duct blockage accident discussed in Section 12.2.10.

iii. Shielding

There is no adverse effect on the function of shielding features of the system as a result of this accident event. The floodwater provides additional shielding that would further reduce radiation dose.

iv. Criticality

There is no adverse effect on the criticality control features of the stored MPC as a result of this accident event. The criticality analysis is unaffected because under the flooding condition water cannot enter the MPC contents space and therefore the reactivity would be less than that under the loading condition in the spent fuel pool (when the MPC is flooded).

v. Confinement

Because, as shown in Chapter 3, the external pressure on the MPC from the DBF is well below its design basis pressure bearing capacity, there is no risk of degradation of the confinement function of the MPC as a result of this accident event. As discussed in the structural evaluation above, all stresses remain within allowable values, assuring confinement boundary integrity.

vi. Radiation Protection and Consequences

Since there is no effect on shielding or confinement functions as discussed above, there is no radiological consequence (from effluents and direct radiation) as a result of this accident event. A minor increase to occupational exposures for the performance of corrective actions is expected.

12.2.4.3 Flood Accident Corrective Action

The configuration of the HI-STORM UMAX VVMs makes them uniquely suited to withstand a flooding event. Indeed, introducing water in the CEC is an effective method to lower the MPC contents' temperature. However, accumulation of debris in the intake plenum or the storage cavities is undesirable as is the risk of corrosion from long-term exposure to floodwaters. Thus, while the short-term effect of flood on the loaded HI-STORM UMAX VVM is essentially benign, corrective actions after such an event are necessary. Visual examination using a boroscope or a camera or temperature monitoring of the exiting air to identify blockage of the

cooling passages following flooding or other site specific natural events is necessary to ensure adequate cooling.

If a state of vent blockage is discovered, then corrective actions to alleviate such condition will be required. To restore the system to a normal configuration, all floodwater and any debris deposited by the receding water must be removed. The specific methods to be used shall be addressed in the site emergency action plan. Examples of acceptable cleaning approaches include:

1. The MPC is removed from the VVM using the HI-TRAC VW transfer cask, allowing direct access to the interior of the VVM through both the inlet vents and the top of the module cavity. Water sprays and vacuuming are used to directly clean the VVM passages and surfaces.

2. Appropriate vacuuming equipment is inserted through the inlet plenum and down to the transverse shells. Water is sprayed in through the outlet vents. Remote cameras are used to inspect the VVM cooling passages to identify and remove debris.

The adequacy of the cooling passages clearance operation is verified by visual inspection or, if the optional temperature monitoring is used, the return of the control temperatures to within allowable limits.

12.2.4.4 Conclusion

Based on the above evaluation, it is concluded that the flood accident does not affect the safe operation of the loaded HI-STORM UMAX VVMs.

12.2.5 Earthquake (Load Case 03 in Section 2.4)
12.2.5.1 Cause of Event

Earthquake is a terrestrial instability event cause by relative movements in the mantle of the earth. The extent of seismic motion at an ISFSI location is established by geotechnical analyses. The intensity of the earthquake is substantially affected by the time span within which its probability is prognosticated.

12.2.5.2 Analysis of the Effect of Design Basis Earthquake (DBE)

The HI-STORM UMAX storage system has been qualified to a fully articulated DBE event set down in Section 2.4. The specified seismic event for the candidate ISFSI must satisfy the acceptance criteria defined in Chapters 2 and 3 to permit the HI-STORM UMAX ISFSI to be established at the site.

i. Structural

The most significant structural effect of the earthquake on the HI-STORM UMAX system are those corresponding to interface forces arising from the MPC rattling action, which acts to apply stresses on the MPC shell and MPC fuel basket panels. The MPC primary stresses resulting from the earthquake event are confirmed to be bounded by the applicable stress limits, and MPC shell secondary stresses local to the area of MPC lid are limited to small plastic deformation without risk of breaching the MPC shell; therefore, there is no effect on structural function.

All other effects correspond to ability of the VVM CEC, ISFSI Pad, and Support Foundation Pad to resist the earthquake loadings. Because the VVM is buried in the substrate, tip-over of the VVM is not credible. The entire VVM can move laterally with the surrounding and supporting substrate. Because of its underground construction, the HI-STORM UMAX VVM is inherently safe under seismic events. Analyses presented in Chapter 3 show that the VVM will continue to render its intended function under a seismic event whose severity is bounded by the Design Basis Earthquake set forth in Chapter 2. Therefore, there is no adverse effect on the structural function of the system.

ii. Thermal

There is no effect on the function of HI-STORM UMAX VVM heat transfer features as a result of this accident event because no constriction of the air flow passages within the system is predicted to occur. Concentricity between the MPC within the CEC shell is maintained by design features and therefore the effect of MPC movement within the storage cavity has a negligible impact on air flow distribution. Thus, the cooling effectiveness of the HI-STORM UMAX VVMs is expected to remain essentially undiminished in the wake of a DBE event.

iii. Shielding

There is no adverse effect on the function of shielding features of the system as a result of this accident event.

iv. Criticality

There is no effect on the criticality control features of the MPC as a result of this accident event.

v. Confinement

There is no effect on the confinement function of the MPC as a result of this accident event. Structural evaluation shows all stresses and strains do not exceed design criteria, assuring confinement boundary integrity.

vi. Radiation Protection and Consequences

Since there is no effect on shielding or confinement functions as discussed above, there is no radiological consequence (from effluents and direct radiation) as a result of this accident event. A minor increase to occupational exposures for the performance of corrective actions is expected.

12.2.5.3 Earthquake Accident Corrective Action

Under a seismic event at an ISFSI, any damage to the HI-STORM UMAX system is expected to be localized and limited to the MPC guides and the MPC shell. A visual inspection in the wake of a seismic event shall be performed as follows:

- Visual inspection to confirm the extent of damage (if any) to the MPC shell is negligible.
- Visual inspection to verity the extent of damage (if any) to other VVM components important-to-safety is negligible.
- Visual inspection to confirm that the insulation attached to the Divider shell remains securely attached and undamaged
- Visual inspection to confirm all air flow passages are clear of obstructions.

Inspections requirements may be modified depending on the severity of the earthquake and other site-specific conditions. Corrective actions shall be implemented based on the results of the inspection.

12.2.5.4 Conclusion

Based on the above evaluation, it is concluded that the Design Basis Earthquake will not affect the safe operation of the HI-STORM UMAX system. Corrective actions, however, may be necessary to restore the system to the pre-seismic condition.

12.2.6 100% Fuel Rod Rupture

This accident event postulates the non-mechanistic condition that all the fuel rods rupture and that the quantities of fission product gases and fill gas are released from the fuel rods into the MPC cavity consistent with ISG-5, Revision 1.

CERTIFICATE OF SERVICE

I hereby certify that on July 20, 2020, I electronically filed the foregoing with the Clerk of the Court for the United States Court of Appeals for the Ninth Circuit by using the appellate CM/ECF system.

Participants in the case who are registered CM/ECF users will be served by the appellate CM/ECF system.

Dated: July 20, 2020

Respectfully submitted,
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No. 20-70899

**IN THE UNITED STATES COURT OF APPEALS
FOR THE NINTH CIRCUIT**

IN RE PUBLIC WATCHDOGS,

Petitioner,

v.

UNITED STATES NUCLEAR REGULATORY COMMISSION,

Respondent.

SOUTHERN CALIFORNIA EDISON COMPANY,
Intervenor.

**INTERVENOR SOUTHERN CALIFORNIA EDISON'S
SUPPLEMENTAL EXCERPTS OF RECORD**

VOLUME 7 OF 8

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SUPPLEMENTAL EXCERPTS OF RECORD:

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12.2.6.1 Cause of 100% Fuel Rod Rupture

Through all credible accident conditions, the HI-STORM UMAX system maintains the spent nuclear fuel in an inert environment while maintaining the peak fuel cladding temperature below the required short-term temperature limits, thereby providing assurance of fuel cladding integrity. Therefore, there is no credible cause for 100% fuel rod rupture. This accident is postulated in NUREG-1536 to evaluate the MPC confinement barrier for the maximum possible internal pressure based on the *non-mechanistic* failure of 100% of the fuel rods.

12.2.6.2 Analysis

The following is an evaluation of effects on structural, thermal, criticality, confinement, and radiation protection performance on the HI-STORM UMAX storage system.

i. Structural

The effect of 100% rod rupture on the MPC is an increase in pressure boundary stresses. Calculations in Chapter 4 show that the accident internal pressure limit bounds the pressure from the 100% fuel rod rupture event; therefore, there is no effect on the MPC's structural function.

ii. Thermal

A bounding MPC internal pressure for the 100% fuel rod rupture condition is presented in Table 4.4.7. The design basis accident condition MPC internal pressure set in Table 2.3.5 and used in the structural evaluation bounds the calculated value. The increased pressure due to the 100% rod rupture has the concomitant (beneficial) effect of enhanced heat transfer through the gases in the MPC cavity. It is concluded that temperatures and pressures resulting from the accident event are bounded by the applicable system temperature and pressure limits; therefore, there is no adverse effect on thermal function.

v. Shielding

There is no adverse effect on function of the shielding features of the system as a result of this accident event.

vi. Criticality

There is no effect on the function of criticality control features of the MPC as a result of this accident event.

vii. Confinement

There is no effect on the function of confinement features of the MPC as a result of this accident event. Structural evaluation shows all stresses remain within allowable values, assuring confinement boundary integrity.

viii. Radiation Protection and Consequences

Since there is no effect on shielding or confinement functions as discussed above, there is no radiological consequence (from effluents and direct radiation) and no expected increase to occupational exposures as a result of this accident event.

Based on the above, it is concluded that the non-mechanistic 100% fuel rod rupture accident event does not affect the safe operation of the HI-STORM UMAX system.

12.2.6.3 100% Fuel Rod Rupture Dose Calculations

The breach of fuel cladding postulated in this accident event does not result in any physical change to the storage system other than some release of gases and a limited quantity of solids (particulates) into the gaseous helium space. The amount of the radiation source remains unaffected. Hence, the radiation dose at the site boundary will not change perceptibly, i.e., there are no consequences to the site boundary dose.

12.2.6.4 100% Fuel Rod Rupture Accident Corrective Action

As shown in the analysis of the 100% fuel rod rupture accident, the MPC Confinement Boundary is not damaged. The HI-STORM UMAX storage System is designed to withstand this accident and continue performing the safe storage of spent nuclear fuel under normal storage conditions. No corrective actions are required.

12.2.6.5 Conclusion

The above evaluation shows that this accident event will not adversely affect the continued safety of the storage system

12.2.7 Confinement Boundary Leakage

None of the postulated environmental phenomenon or accident conditions identified in Chapter 2 has been determined to precipitate failure of the confinement boundary. The MPC uses redundant confinement closures to assure that there is no release of radioactive materials. The analyses presented in the HI-STORM FW FSAR and in Chapter 3 herein demonstrate that the MPC remains intact during all postulated accident conditions. The information contained in Chapter 7 of the HI-STORM FW FSAR demonstrates that the MPC is designed, fabricated, tested, and inspected to meet the guidance of ISG-18 such that unacceptable leakage from the Confinement Boundary is non-credible.

12.2.8 Explosion (Load Case 01 in Section 2.4)

12.2.8.1 Cause of Explosion

An explosion within the protected area of an ISFSI is improbable since there are no explosive materials permitted within the site boundary. However, an explosion as a result of combustion of the fuel contained in a cask transport vehicle is possible. As the fuel available for the explosion is limited in quantity, the effects of an explosion on a reinforced structure are minimal. Explosions that are credible for a specific ISFSI would require a site hazards evaluation under the provisions of 10CFR72.212 regulations by the ISFSI owner using the methodology set forth in Chapter 3.

12.2.8.2 Explosion Analysis

Any credible explosion accident for the MPC is bounded by the accident external design pressure (Table 2.3.5). Because explosive materials are not stored within close proximity to the casks, the design basis pressure wave from explosion is limited to a small value (Table 2.3.1). The bounding analysis in Chapter 3 shows that the MPC and the CEC can withstand the effects of substantial accident external pressures without collapse or rupture.

i. Structural

The effect of explosion at the HI-STORM UMAX ISFSI is a near instantaneous increase of external pressure over the top exterior surface of the VVM closure lid and in the air passages and cavity space of the VVM due to the explosion-induced pressure wave. Chapter 3 includes an evaluation of the effect of the design-basis pressure wave set in Table 2.3.1 and applied as a static pressure on the exterior surfaces of the closure lid producing a downward force. Since the pressure wave entering the VVM through the closure lid vents will have slightly less energy, there is no need to consider uplift of the closure lid. Thus, the evaluation shows that the overpressure wave does not result in lid separation and that all lid stresses are a fraction of the allowable limits. Therefore, the continued structural integrity of the Closure Lid is assured.

Site-specific explosion scenarios that are not evidently bounded by the design basis explosion load considered in this FSAR shall be evaluated under the provisions of 10CFR72.212.

ii. Thermal

There is no effect on the function of HI-STORM UMAX VVM heat transfer features as a result of this accident event occurring at the ISFSI. No deformation of the HI-STORM UMAX VVM components that would result in the constriction of the air flow passages within the VVM is indicated.

iii. Shielding

There is no effect on the function of shielding features of the system as a result of this accident event.

v. Criticality

There is no effect on the function of criticality control features of the MPC as a result of this accident event.

vi. Confinement

There is no effect on the confinement function of the MPC as a result of this accident event. As the above mentioned structural evaluation shows, all stresses remain within allowable values, assuring confinement boundary integrity.

vii. Radiation Protection and Consequences

Since there is no effect on shielding or confinement functions as discussed above, there is no radiological consequence (from effluents and direct radiation) as a result of this accident event. A negligible-to-minor increase to occupational exposures for the performance of corrective actions is expected.

12.2.8.3 Corrective Action

As there is no permanent damage indicated by this accident event, there is no need for a corrective action.

12.2.8.4 Conclusion

Based on the above evaluation, it is concluded that the design basis explosion accident event does not affect the safe operation of the loaded HI-STORM UMAX storage system.

12.2.9 Lightning

12.2.9.1 Cause of Lightning

Lightning is a meteorological event that occurs in all parts of the world.

12.2.9.2 Lightning Analysis

Because of its underground construction, the subterranean portion of the HI-STORM UMAX System is unlikely to be subjected to a direct lightning strike. The HI-STORM UMAX closure lid is, however, aboveground (albeit a low profile structure) and could be subjected to a direct strike.

When the HI-STORM UMAX VVM is struck with lightning, the lightning will discharge through the steel shell of the lid and the CEC structure to the ground. Lightning strikes have high currents, but their duration is short (i.e., less than a second). The VVM shell and lid are composed of conductive steel and, as such, provides a direct path to the ground into the substrate with which it has a large interface.

Because the VVMs are buried in substrate, they are self-grounding. The lightning current will discharge into the VVM steel structure and directly into the ground. Therefore, the MPC (made of relatively non-conductive austenitic stainless steel) will be unaffected.

i. Structural

There is no structural consequence as a result of this event.

ii. Thermal

There is no effect on the thermal performance of the system as a result of this event.

iii. Shielding

There is no effect on the shielding performance of the system as a result of this event.

iv. Criticality

There is no effect on the criticality control features of the system as a result of this event.

v. Confinement

There is no effect on the confinement function of the MPC as a result of this event.

vi. Radiation Protection

Since there is no degradation in shielding or confinement capabilities as discussed above, there is no effect on occupational or public exposures as a result of this event.

12.2.9.3 Lightning Dose Calculations

As lightning strike has no effect on the Confinement Boundary or shielding materials, no dose analysis is necessary.

12.2.9.4 Lightning Accident Corrective Action

The HI-STORM UMAX System will not sustain any damage from the lightning accident that might adversely affect its performance. Therefore, no surveillance or corrective action is required subsequent to a lightning action at the HI-STORM UMAX ISFSI.

12.2.9.5 Conclusion

Based on this evaluation, it is concluded that a lightning event will not affect the safe operation of the HI-STORM UMAX System.

12.2.10 100% Blockage of Air Inlet

12.2.10.1 Cause of 100% Blockage of Air Inlet

This event is defined as a complete blockage of all VVM inlets. A complete blockage of all VVM inlets cannot be realistically postulated to occur at most sites. However, a flood, blizzard snow accumulation, tornado debris, or volcanic activity, where applicable, can cause a significant blockage.

12.2.10.2 100% Blockage of Air Inlet Analysis

The immediate consequence of a complete blockage of the air inlet openings is that the normal circulation of air for cooling the MPC is stopped. An amount of heat will continue to be removed by localized air circulation patterns in the outlet opening, and the MPC will continue to radiate heat to the relatively cooler soil. As the temperatures of the MPC and its contents rise, the rate of heat rejection will increase correspondingly. Under this condition, the temperatures of the HI-STORM UMAX overpack, the MPC and the stored fuel assemblies will rise as a function of time.

As a result of the large mass and correspondingly large thermal capacity of the HI-STORM UMAX overpack, it is expected that a significant temperature rise is only possible if the blocked condition is allowed to persist for an extended duration. This accident condition is, however, a short duration event that will be identified by the ISFSI staff, at worst, during scheduled periodic surveillance at the ISFSI site and corrected using the site's emergency response process.

i. Structural

There are no structural consequences as a result of this event

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ii. Thermal

A thermal analysis is performed in Subsection 4.6.2.3 to determine the effect of a complete blockage of all inlets for an extended duration. For this event, both the fuel cladding and

component temperatures (see Table 4.6.7) remain below their accident temperature limits (see Table 2.3.7). The MPC internal pressure for this event is evaluated and reported in Table 4.6.10 and is bounded by the design basis internal pressure for accident conditions (see Table 2.3.5).

iii. Shielding

The above thermal results indicate insignificant loss of material and, therefore, the effect of this event on the shielding capacity is expected to be negligible.

iv. Criticality

There is no effect on the function of criticality control features of the MPC as a result of this accident event.

v. Confinement

There is no effect on the confinement function of the MPC as a result of this accident event.

vi. Radiation Protection and Consequences

Since there is no effect on shielding or confinement functions as discussed above, there is no radiological consequence (from effluents and direct radiation) as a result of this event. A negligible-to-minor increase to occupational exposures for the performance of corrective actions is expected.

12.2.10.3 Corrective Action

Analysis of the 100% blockage of air inlet accident shows that the temperatures for system components and fuel cladding are within the accident temperature limits if the blockage is cleared within the maximum elapsed period between scheduled surveillance inspections. Upon detection of the complete blockage of the air inlet openings, the ISFSI owner shall activate its emergency response procedure to remove the blockage with mechanical and manual means as necessary. After clearing the overpack openings, the system shall be visually and radiologically inspected for any damage. If exit air temperature monitoring is performed in lieu of direct visual inspections, the difference between the ambient air temperature and the exit air temperature will be the basis for the assurance that the temperature limits are not exceeded.

For an accident event that completely blocks the inlet or outlet air openings for greater than the analyzed duration, a site-specific evaluation or analysis may be performed to whether adequate heat removal for the duration of the event would occur. Adequate heat removal is defined as the minimum rate of heat dissipation that ensures cladding temperatures limits are met and structural integrity of the MPC and overpack is not compromised. For those events where an evaluation or analysis is not performed or is not successful in showing that cladding temperatures remain

below their short term temperature limits, the site's emergency plan shall include provisions to address removal of the material blocking the air inlet openings and to provide alternate means of cooling prior to exceeding the time when the fuel cladding temperature reaches its short-term temperature limit. Alternate means of cooling could include, for example, spraying water into the air outlet opening using pumps or fire-hoses or blowing air into the air outlet opening, to directly cool the MPC.

12.2.10.4 Conclusion

Based on the above evaluation, it is concluded that the 100% blockage of air inlet accident event does not affect the safe operation of the HI-STORM UMAX System, if the blockage is removed in the specified time period.

12.2.11 Burial Under Debris

12.2.11.1 Cause of Burial Under Debris

Complete burial of the entire HI-STORM UMAX VVM assembly is not a credible accident because it is a large structure (see the drawing package in Section 1.5) and during storage at the ISFSI, there are no large structures above the casks that may collapse and bury the VVM. The minimum regulatory distance(s) from the ISFSI to the nearest site boundary and the controlled area around the ISFSI concrete pad precludes the close proximity of substantial amounts of vegetation. However, for purposes of safety evaluation, complete burial of the VVM including blockage of all inlet and outlet flow passages is assumed.

12.2.11.2 Burial Under Debris Analysis

Burial of the inlet plenum under debris will adversely affect thermal performance because the debris will block the inflow of air. This will cause the fuel cladding temperatures to increase. A thermal analysis has been performed to determine the time for the fuel cladding temperatures to reach the *accident condition temperature limit* due to a burial of the inlet plenum under debris accident, assuming that the debris causes complete cut-off of cool air through the inlet passages. This computed time, T-max, is specified in the Technical Specification as an LCO.

i. Structural

The effect of 100% blockage of air inlet on the MPC is an increase in component and fuel cladding temperatures and internal pressure and thus an increase in pressure boundary stresses. However, the resultant temperatures and pressures, for a burial duration equal to T-max, are below the accident condition design limits as discussed in the thermal effects evaluation below. The MPC stresses resulting from the blockage of air inlet are confirmed to be bounded by the applicable ASME code limits; therefore, there is no effect on structural function.

ii. Thermal

The fuel cladding and MPC integrity is evaluated in Subsection 4.6.2.4. The evaluation demonstrates that the fuel cladding and confinement function of the MPC are not compromised even if the burial event lasts for a substantial duration.

iii. Shielding

The above thermal results indicate that there will be no material loss in the shielding capacity of the system.

iv. Criticality

There is no effect on the function of criticality control features of the MPC as a result of this accident event.

v. Confinement

There is no effect on the confinement function of the MPC as a result of this accident event.

vi. Radiation Protection and Consequences

Since there is no effect on shielding or confinement functions as discussed above, there is no radiological consequence (from effluents and direct radiation) as a result of this event. A negligible-to-minor increase to occupational exposures for the performance of corrective actions is expected.

12.2.11.3 Corrective Action

Analysis of the burial-under-debris accident shows that the fuel cladding peak temperatures are not exceeded for the duration of the accident equal to T-max. Upon detection of the burial-under-debris accident, the ISFSI operator shall assign personnel to remove the debris from around and inside the VVM cavity with mechanical and manual means as necessary. After removing the debris, the storage cavities shall be visually inspected for any damage. The loaded MPC may have to be removed from the VVM cavities to allow complete inspection of the VVM cavities. Removal of obstructions to the air flow path shall be performed prior to the re-insertion of the MPC. The site's emergency action plan shall include provisions for the implementation of this corrective action.

12.2.11.4 Conclusion

Based on the above evaluation, it is concluded that the burial-under-debris accident event does not affect the safe operation of the HI-STORM UMAX System, if the blockage is removed in the specified time period.

12.2.12 Extreme Environmental Temperature

12.2.12.1 Cause of Extreme Environmental Temperature

The extreme environmental temperature is postulated (see Table 2.3.6) as a 3-day average temperature caused by extreme weather conditions.

12.2.12.2 Extreme Environmental Temperature Analysis

To determine the effects of the extreme temperature, it is conservatively assumed that the temperature persists for a sufficient duration (3 days) to allow the HI-STORM storage system to achieve thermal equilibrium.

The accident condition considering an extreme environmental temperature (Table 2.3.6) for a duration sufficient to reach thermal equilibrium is evaluated with respect to accident condition design temperatures listed in Table 2.3.7.

i. Structural

The effect on the MPC for the extreme environmental temperature conditions is an increase in component and fuel cladding temperatures and internal pressure and thus an increase in pressure boundary stresses. However, the resultant temperatures and pressures are below the accident design limits as discussed in the thermal effects evaluation below. The MPC stresses resulting from the extreme environmental temperatures event are confirmed to be bounded by the applicable pressure boundary stress limits; therefore, there is no effect on structural function.

ii. Thermal

The thermal evaluation of extreme environmental temperature is discussed in Section 4.6.2.2. The calculated bounding temperatures and pressures are conservatively evaluated assuming the asymptotic maximum (steady-state) condition to have been reached. Temperature results reported in Table 4.6.6 indicate that all thermal criteria are met. Additionally, the increased temperatures generate an elevated MPC internal pressure, reported in Table 4.6.10, which is less than the accident design pressure limit specified in Table 2.3.5. The temperatures and pressures resulting from the increased pressure event are confirmed to be bounded by the applicable system temperature and pressure limits; therefore, there is no effect on thermal function.

iii. Shielding

There is no effect on the function of shielding features of the system as a result of this accident event. Concrete temperatures in the lid are confirmed to remain below its accident temperature limit.

iv. Criticality

There is no effect on the function of criticality control features of the MPC as a result of this accident event.

v. Confinement

There is no effect on the confinement function of the MPC as a result of this accident event. Structural evaluation shows all stresses remain within allowable values, assuring confinement boundary integrity.

vi. Radiation Protection and Consequences

Since there is no effect on shielding or confinement functions as discussed above, there is no radiological consequence (from effluents and direct radiation) and no increase to occupational or public exposures as a result of this accident event.

12.2.12.3 Corrective Action

The extreme environmental temperature is a self-correcting event. No corrective action is required.

12.2.12.4 Conclusion

Based on this evaluation, it is concluded that the extreme environment temperature accident event does not affect the safe operation of the HI-STORM UMAX System.

12.2.13 HI-TRAC VW Transfer Cask Handling Accident

HI-TRAC VW transfer cask handling accident evaluated in the HI-STORM FW FSAR Section 12.2.1 [4.1.2] is incorporated by reference.

12.3 OTHER EVENTS

This section addresses miscellaneous events, which are placed in the category of "other events" since they cannot be categorized as off-normal or accident events. The following "other events" are discussed in this chapter:

- Hazards during Construction Proximate to the ISFSI
- MPC Reflood

The results of the evaluations performed herein demonstrate that the loaded HI-STORM UMAX VVMs can withstand the effects of "other events" without affecting safety function.

12.3.1 Construction Proximate to an Operating ISFSI

12.3.1.1 Cause of the Event

This situation will arise if the facility owner decides to expand the existing storage capacity by adding VVM assemblies adjacent to an operating ISFSI. As demonstrated in Chapter 3, the Self-hardening Engineered Subgrade (SES), in the absence of the optional Enclosure Wall, can remain intact under the Design Basis Earthquake condition and can effectively protect loaded VVMs against tornado missile impacts. This FSAR permits excavation adjacent to one side of the HI-STORM UMAX ISFSI at any given time. However, the excavation depth shall be limited to the bottom surface of the Support Foundation Pad (SFP).

12.3.1.2 Safety Analysis

i. Structural

The soil-structure interaction analysis of the ISFSI assuming that the subgrade has been excavated down to the SFP elevation for (a theoretically) infinite distance next to one side of the ISFSI has been performed. The analyses show that the storage system is kinematically stable and the stresses in the reinforced concrete structures meet the applicable ACI limits and that the canister retrievability in the wake of an earthquake is assured. In the absence the optional Enclosure Wall, the tensile stress developed in the SES under the Design Basis Earthquake condition is well below the tensile capacity of the SES material. Moreover, the local strains in the stored canisters from internal impact are well within the limit specified in this FSAR.

Furthermore, the scenario of a tornado missile striking the exposed SES has also been considered in Chapter 3. Analyses summarized in Section 3.4 show that the design basis projectiles (large, medium, or small), specified in Chapter 2 of this FSAR, applied in the most vulnerable location of the construction cavity, will fail to reach any stored MPC.

ii. Thermal

There is no effect on the thermal performance of the system as a result of this event.

vii. Shielding

There is no effect on the function of shielding features of the system as a result of this event.

viii. Criticality

There is no effect on the function criticality control features of the MPC as a result of this event.

ix. Confinement

There is no effect on the function of confinement features of the MPC as a result of this event.

x. Radiation Protection

To protect an installed ISFSI from any site construction activity in its proximity, a certain minimum ground buffer distance beyond the edge of the perimeter of the VVM arrays is prescribed in the licensing drawings. This radiation protection space (RPS) defines the no-construction zone around the installed and loaded VVMs (see Chapter 1).

A generic evaluation of the shielding consequences of digging a cavity adjacent to the RPS has been considered in Chapter 5 of this FSAR. The analyses show that the dose at the edge of the cavity is well below 0.2 mrem/hr, which is well below the customary limit that requires radiation posting at nuclear power plants. Therefore, the excavation activities shall be ALARA.

Nevertheless, the owner will implement appropriate measures and provide appropriate safety training to the construction crew in keeping with the plant's radiation protection plan. Analysis of the consequences of any credible site-specific loads or events during site construction work shall be performed with due consideration of the duration and nature of the site construction activity. As is required for deploying casks certified under 10CFR72, Subpart L, every site modification that may potentially impact the continued operability of the ISFSI must be evaluated for acceptability under 10CFR72.212.

Because the actual projectiles for a specific ISFSI site are often different from the tornado-borne missiles analyzed in Chapters 3 and 5 herein, a site-specific analysis of the effect of all credible missiles shall be performed assuming that the largest construction cavity adjacent to the ISFSI

exists. PRA considerations shall not be used to rule out any missile that has been determined to be credible in the plant's FSAR.

To summarize, the RPS provides sufficient margin (buffer) against design basis projectiles analyzed in Chapter 3. In addition to the generic analyses documented in this FSAR, a site-specific evaluation pursuant to §72.212 shall be performed for all other credible hazards that can be postulated during site construction. Administrative controls to guard against accidental human error in excavations (such as encroachment of the RPS) shall be addressed through written procedures consistent with the required controls needed for a safety significant activity within a Part 50 controlled area.

Furthermore, the ISFSI owner shall implement ameliorative measures to prevent unacceptable damage to the ISFSI from any other credible adverse scenarios unique to a site that has not been considered in this FSAR. An example of such a measure is the installation of a berm to protect against environmental events such as soil erosion and mud slides. Such site-specific design initiatives at any "UMAX" ISFSI, like its aboveground counterpart, are within the purview of the plant's §72.212 process.

12.3.1.3 Corrective Action

As the excavation work is a planned activity, no corrective action is required.

12.3.1.4 Conclusion

An excavation activity adjacent to an operating ISFSI is permitted provided all safety and radiation protection requirements of the host plant are followed.

12.3.2 MPC Reflood

MPC reflood evaluated in the HI-STORM FW FSAR Section 12.3.1 [4.1.2] is incorporated by reference.

CHAPTER 13: OPERATING CONTROLS AND LIMITS

13.0 INTRODUCTION

This chapter defines the operating controls and limits (i.e., Technical Specifications) including their supporting bases for deployment and storage of an approved MPC type (see Table 1.2.1) in a HI-STORM UMAX VVM at an ISFSI. Table 1.2.1 provides the CoC amendment numbers for the MPCs that have been considered in the safety analysis documented in this report. Thus while the safety analyses have been carried out in this FSAR for all MPCs presently certified in Docket # 72-1014 (HI-STORM 100) and Docket # 1032 (HI-STORM UMAX), the Certificate-of- Compliance sought pursuant to this licensing submittal is limited to qualifying only MPC-37 and MPC-89 which have been previously certified in the HI-STORM FW docket.

13.1 PROPOSED OPERATING CONTROLS AND LIMITS

13.1.1 NUREG-1536 (Standard Review Plan) Acceptance Criteria

This portion of the FSAR establishes the commitments regarding the HI-STORM UMAX system and its use. Other 10CFR72 [13.1.1] and 10CFR20 [13.1.2] requirements in addition to the Technical Specifications may apply. The conditions for a general license holder found in 10CFR72.212 [13.1.1] shall be met by the licensee prior to loading spent fuel into the HI-STORM UMAX system. The general license conditions governed by 10CFR72 [13.1.1] are not repeated within these Technical Specifications. Licensees are required to comply with all commitments and requirements.

The Technical Specifications provided in Appendix A to the CoC and the authorized contents and design features provided in Appendix B to the CoC are primarily established to maintain subcriticality, the confinement boundary, shielding and radiological protection, heat removal capability, and structural integrity under normal, off-normal and accident conditions.

Table 13.1.1 addresses each of these conditions applicable to HI-STORM UMAX and identifies the appropriate Technical Specification(s) designed to control the condition Table 13.1.2 provides the list of Technical Specifications for the loading operations and long term fuel storage in the HI-STORM UMAX system.

Table 13.1.1

HI-STORM UMAX SYSTEM CONTROLS

Condition to be Controlled	**Applicable Technical Specifications**†
Criticality Control	3.3.1 Boron Concentration
Confinement boundary integrity and integrity of cladding on undamaged fuel	3.1.1 Multi-Purpose Canister (MPC)
Shielding and radiological protection	3.1.1 Multi-Purpose Canister (MPC) 3.1.3 MPC Reflooding 3.2.1 TRANSFER CASK Surface Contamination 5.1 Radioactive Effluent Control Program 5.3 Radiation Protection Program
Heat removal capability	3.1.1 Multi-Purpose Canister (MPC) 3.1.2 SFSC Heat Removal System
Structural integrity	5.2 Transport Evaluation Program

† Technical Specifications are located in Appendix A to the CoC. Authorized contents are specified in this FSAR in Subsection 2.1.8

Table 13.1.2

HI-STORM UMAX SYSTEM TECHNICAL SPECIFICATIONS

NUMBER	**TECHNICAL SPECIFICATION**
1.0	USE AND APPLICATION
1.1	DEFINITIONS
1.2	LOGICAL CONNECTORS
1.3	COMPLETION TIMES
1.4	FREQUENCY
2.0	Not Used
3.0	LIMITING CONDITION FOR OPERATION (LCO) APPLICABILITY SURVEILLANCE REQUIREMENT (SR) APPLICABILITY
3.1	SFSC Integrity
3.1.1	Multi-Purpose Canister (MPC)
3.1.2	SFSC Heat Removal System
3.1.3	MPC Cavity Reflooding
3.2	SFSC Radiation Protection
3.2.1	TRANSFER CASK Surface Contamination
3.3	SFSC Criticality Control
3.3.1	Boron Concentration
Table 3-1	MPC Cavity Drying Limits
Table 3-2	MPC Helium Backfill Limits
4.0	Not Used
5.0	ADMINSTRATIVE CONTROLS
5.1	Radioactive Effluent Control Program
5.2	Transport Evaluation Program
5.3	Radiation Protection Program

13.2 DEVELOPMENT OF OPERATING CONTROLS AND LIMITS

This section provides a discussion of the operating controls and limits, and training requirements for the HI-STORM UMAX system to assure long-term performance consistent with the conditions analyzed in this FSAR.

13.2.1 Training Modules

Training modules are to be developed under the licensee's training program to require a comprehensive, site-specific training, assessment, and qualification (including periodic re-qualification) program for the operation and maintenance of the HI-STORM UMAX Spent Fuel Storage Cask (SFSC) System and the Independent Spent Fuel Storage Installation (ISFSI). The training modules shall include the following elements, at a minimum:

1. HI-STORM UMAX System Design (overview);

2. ISFSI Facility Design (overview);

3. Systems, Structures, and Components Important-to-Safety (overview);

4. HI-STORM UMAX System Safety Analysis Report (overview);

5. NRC Safety Evaluation Report (overview);

6. Certificate of Compliance conditions;

7. HI-STORM UMAX Technical Specifications, Approved Contents, Design Features and other Conditions for Use;

8. HI-STORM UMAX Regulatory Requirements (e.g., 10CFR72.48, 10CFR72, Subpart K, 10CFR20, 10CFR73);

9. Required instrumentation and use;

10. Operating Experience Reviews

11. HI-STORM UMAX System and ISFSI Procedures, including
    - Procedural overview
    - Fuel qualification and loading
    - MPC /HI-TRAC/VVM Closure Lid rigging and handling, including safe load pathways
    - MPC welding operations

- HI-TRAC/VVM staging operation
- Auxiliary equipment operation and maintenance (e.g., draining, moisture removal, helium backfilling and cooldown)
- MPC/HI-TRAC/VVM pre-operational and in-service inspections and tests
- Transfer and securing of the loaded HI-TRAC onto the transport vehicle
- Movement of HI-TRAC to ISFSI.
- Transfer of MPC to VVM
- Preparation of MPC/HI-TRAC VVM for fuel unloading
- Retrieval of MPC from VVM
- Surveillance
- Radiation protection
- Maintenance
- Security
- Off-normal and accident conditions, responses, and corrective actions

13.2.2 Dry Run Training

A dry run training exercise of the loading, closure, handling, and transfer of the HI-STORM UMAX system shall be conducted by the licensee prior to the first use of the system to load spent fuel assemblies. Dry run training already performed successfully for the HI-STORM 100 System can be substituted for dry run steps applicable to HI-STORM UMAX. The dry run shall include, but is not limited to the following:

1. Inspection of the HI-STORM UMAX System components.

2. Moving the MPC/HI-TRAC into the spent fuel pool.

3. Preparation of the MPC/HI-TRAC for fuel loading.

4. Selection and verification of specific fuel assemblies to ensure conformance.

5. Locating specific assemblies and placing assemblies into the MPC/HI-TRAC (using a dummy fuel assembly), including appropriate independent verification.

6. Remote installation of the MPC lid and removal of the MPC/HI-TRAC from the spent fuel pool.

7. MPC welding, NDE inspections, pressure testing, draining, moisture removal, and helium backfilling (for which a mockup MPC may be used).

8 Movement of the loaded HI-TRAC to the ISFSI.

9. Transfer of the MPC from the HI-TRAC into the HI-STORM UMAX VVM at the ISFSI.

13.2.3 Functional and Operating Limits, Monitoring Instruments, and Limiting Control Settings

The controls and limits apply to operating parameters and conditions which are observable, detectable, and/or measurable. The HI-STORM UMAX system is completely passive during storage and requires no monitoring instruments. The user may choose to implement a temperature monitoring system or visually inspect the vent screens to verify operability of the VVM heat removal system in accordance with Technical Specification Limiting Condition for Operation (LCO) 3.1.2.

13.2.4 Limiting Conditions for Operation (LCO)

Limiting Conditions for Operation (LCO) specify the minimum capability or level of performance that is required to assure that the HI-STORM UMAX system can fulfill its safety functions.

13.2.5 Equipment

The HI-STORM UMAX system and its components have been analyzed for specified normal, off-normal, and accident conditions, including extreme environmental conditions. Analysis has shown that no credible condition or event prevents the HI-STORM UMAX system from meeting its safety function. As a result, there is no threat to public health and safety from any postulated accident condition or analyzed event. When all equipment is loaded, tested, and placed into storage in accordance with procedures developed for the ISFSI, no failure of the system to perform its safety function is expected to occur.

13.2.6 Surveillance Requirements

The analyses show that the HI-STORM UMAX system fulfills its safety functions, provided that the Technical Specifications and the Authorized Contents described in Subsection 2.1.8 are met. Surveillance requirements during loading, unloading, and storage operations are provided in the Technical Specifications.

13.2.7 Design Features

This subsection describes HI-STORM UMAX system design features that are Important to Safety. These features require design controls and fabrication controls. The design features, detailed in this FSAR and in Appendix B to the CoC, are established in specifications and drawings which are controlled through the quality assurance program. Fabrication controls and inspections are in place to ensure that the HI-STORM UMAX system is fabricated in accordance with the licensing drawings in Section 1.5.

13.2.8 MPC

a. Basket material composition, properties, dimensions, and tolerances for criticality control.
b. Canister material mechanical properties for structural integrity of the confinement boundary.
c. Canister and basket material thermal properties and dimensions for heat transfer control.
d. Canister and basket material composition and dimensions for dose rate control.

13.2.9 HI-STORM UMAX VVM

a. HI-STORM UMAX VVM material mechanical properties and dimensions for structural integrity to provide protection of the MPC and shielding of the spent nuclear fuel assemblies during handling and storage operations.
b. HI-STORM UMAX VVM material thermal properties and dimensions for heat transfer control.
c. HI-STORM UMAX VVM material composition and dimensions for dose rate control.

13.2.10 HI-TRAC Transfer Cask

a. HI-TRAC transfer cask material mechanical properties and dimensions for structural integrity to provide protection of the MPC and shielding of the spent nuclear fuel assemblies during loading, unloading and handling operations.
b. HI-TRAC transfer cask material thermal properties and dimensions for heat transfer control.
c. HI-TRAC transfer cask material composition and dimensions for dose rate control.

13.2.11 Verifying Compliance with Fuel Assembly Decay Heat, Burnup, and Cooling Time Limits

The examples below demonstrate how the user of the system can determine if fuel assemblies, including NFH, are acceptable for loading in either the MPC-37 or the MPC-89 in accordance with the allowable decay heat, burnup, and cooling time for the approved contents.

Example 1

In this example, it will be assumed that the MPC-37 is being loaded with array/class 17x17A fuel as shown in Figure 2.1.7 with heat loads from Table 2.1.8. Note that sub-design heat load patterns are excluded in this example.

Table 13.2.1 provides four hypothetical fuel assemblies in the 17x17A array/class that will be evaluated for acceptability for loading in the MPC-37. The decay heat values and the fuel classification in Table 13.2.1 are determined by the user. The other information is taken from the fuel assembly and reactor operating records.

Fuel Assembly Number 1 is acceptable for storage in the cells identified in Table 13.2.3 for MPC-37 (Figure 2.1.7) Heat Load Charts 1 and 3 for "Short", "Standard" and "Long" fuel assemblies (depending on the assembly length) displayed in Figures 2.1.12, 2.1.13, 2.1.16, 2.1.17 and 2.1.18. Fuel Assembly Number 1 is not acceptable for storage in the cells identified in Table 13.2.3 for MPC-37 (Figure 2.1.7) Heat Load Charts 1 and 3, and is not acceptable for storage in MPC-37 (Figure 2.1.7) Heat Load Chart 2 for "Short", "Standard" and "Long" fuel assemblies in Figures 2.1.14 and 2.1.15 because the total heat load of the fuel assembly and the non-fuel hardware exceeds the per storage cell decay heat limit. "Short", "Standard" and "Long" fuel assemblies are defined in Table 2.1.7.

Fuel Assembly Number 2 is not acceptable for loading. Fuel Assembly 2 is limited to the cell locations for DFCs in the MPC-37 (Figure 2.1.1). These cells have a decay heat limits lower than the decay heat of the assembly. This assembly will need additional cooling time (reduction in decay heat) to be acceptable for loading in the MPC-37.

Fuel Assembly Number 3 is acceptable for loading in the cells identified in Table 13.2.4 for MPC-37 Heat Load Charts 1 and 2 for "Short", "Standard" and "Long" fuel assemblies (depending on the fuel assembly height) displayed in Figures 2.1.12, 2.1.13, 2.1.14 and 2.1.15. Fuel Assembly Number 3 is also acceptable for loading in the cells identified in Table 13.2.5 for MPC-37 Heat Load Chart 3 for "Short", "Standard" and "Long" fuel displayed in Figures 16, 17 and 18. The fuel assembly is limited to these locations due to the non-fuel hardware (Figure 2.1.5) and the total heat load of the fuel assembly and non-fuel hardware is less than the decay heat limits for these cells.

Fuel Assembly Number 4 is not acceptable for loading in the MPC-37 because its cooling time is less than the minimum of 3 years. When the fuel assembly attains three years cooling time it can be reevaluated based on the decay heat.

Example 2

In this example, it will be assumed that the MPC-89 is being loaded with array/class 10x10A fuel as shown in Figure 2.1.8 with heat loads from Table 2.1.9. Sub-design heat loads are excluded in this example.

Table 13.2.2 provides four hypothetical fuel assemblies in the 10x10A array/class that will be evaluated for acceptability for loading in the MPC-89. The decay heat values and the fuel classification in Table 13.2.2 are determined by the user. The other information is taken from the fuel assembly and reactor operating records.

Fuel Assembly Number 1 is acceptable for loading in the MPC-89 as Moderate Burnup Fuel (MBF) for long-term storage (Figure 2.1.22). Fuel Assembly 1 is limited to the cell locations for DFCs in the MPC-89 (Figure 2.1.2). These cells have a decay heat limit higher than the decay heat of the assembly, therefore the assembly is acceptable for loading in the MPC-89 as MBF for long-term storage, but it is limited to the cells depicted in Figure 2.1.2.

Fuel Assembly Number 2 is not acceptable for loading in the MPC-89 as MBF (Figure 2.1.22). Fuel Assembly 2 is limited to the cell locations for DFCs in the MPC-89 (Figure 2.1.2). These cells have a decay heat limit lower than the decay heat of the assembly. This assembly will need additional cooling time (reduction in decay heat) to be acceptable for loading in the MPC-89.

Fuel Assembly Number 3 is acceptable for loading in all cell locations of the MPC-89 (Figure 2.1.8) as MBF for long-term storage.

Fuel Assembly Number 4 is not acceptable for loading in the MPC-89 (Figure 2.1.22). The decay heat limit of the cells is lower than the decay heat limit of the assembly.

Table 13.2.1

SAMPLE CONTENTS TO DETERMINE ACCEPTABILITY FOR STORAGE
(Array/Class 17x17A)

FUEL ASSEMBLY NUMBER	**1**	**2**	**3**	**4**
INITIAL ENRICHMENT (WT. % ^{235}U)	3.0	3.2	4.3	4.5
FUEL ASSEMBLY BURNUP (MWD/MTU)	37100	35250	41276	55000
FUEL ASSEMBLY COOLING TIME (YEARS)	4.7	3.3	18.2	2.9
FUEL ASSEMBLY DECAY HEAT (KW)	1.01	1.45	0.4	2.08
NON-FUEL HARDWARE STORED WITH ASSEMBLY	BPRA	None	NSA	None
NFH DECAY HEAT (KW)	0.5	0	0.3	0
FUEL CLASSIFICATION	Undamaged	Damaged	Undamaged	Undamaged

Table 13.2.2

SAMPLE CONTENTS TO DETERMINE ACCEPTABILITY FOR STORAGE
(Array/Class 10x10A)

FUEL ASSEMBLY NUMBER	**1**	**2**	**3**	**4**
INITIAL ENRICHMENT (WT. % ^{235}U)	3.0	3.2	4.3	4.5
FUEL ASSEMBLY BURNUP (MWD/MTU)	37100	35250	41276	55000
FUEL ASSEMBLY COOLING TIME (YEARS)	4.7	3.3	18.2	7
FUEL ASSEMBLY DECAY HEAT (KW)	0.43	0.55	0.2	0.61
FUEL CLASSIFICATION	Damaged	Damaged	Undamaged	Undamaged

Table 13.2.3: Fuel Assembly Number 1 in Table 13.2.1	
MPC-37 cells Authorized for Storage	MPC-37 cells Not Authorized for Storage
5, 6, 7, 10, 14, 17, 21, 24, 28, 31, 32, 33	1, 2, 3, 4, 8, 9, 11, 12, 13, 15, 16, 18, 19, 20, 22, 23, 25, 26, 27, 29, 30, 34, 35, 36, 37

Note: See Subsection 13.2.11, Example 1 for discussion.

Table 13.2.4: Storage of Fuel Assembly Number 3 (Table 13.2.1) in the MPC-37 (Heat Load Charts 1 and 3)	
MPC-37 cells Authorized for Storage	MPC-37 cells Not Authorized for Storage
5, 6, 7, 10, 11, 12, 13, 14, 17, 18, 19, 20, 21, 24, 25, 26, 27, 28, 31, 32, 33	1, 2, 3, 4, 8, 9, 15, 16, 22, 23, 29, 30, 34, 35, 36, 37

Note: See Subsection 13.2.11, Example 1 for discussion.

Table 13.2.5: Storage of Fuel Assembly Number 3 (Table 13.2.1) in MPC-37 (Heat Load Chart 2)	
MPC-37 cells Authorized for Storage	MPC-37 cells Not Authorized for Storage
5, 6, 7, 10, 11, 12, 13, 14, 17, 18, 20, 21, 24, 25, 26, 27, 28, 31, 32, 33	1, 2, 3, 4, 8, 9, 15, 16, 19, 22, 23, 29, 30, 34, 35, 36, 37

Note: See Subsection 13.2.11, Example 1 for discussion.

13.3 TECHNICAL SPECIFICATIONS

Technical Specifications for the HI-STORM UMAX system are provided in Appendix A to the Certificate of Compliance. Authorized Contents (i.e., fuel specifications) and Design Features are provided in Appendix B to the CoC. Bases applicable to the Technical Specifications are provided in the FSAR Appendix 13.A. The format and content of the HI-STORM UMAX system Technical Specifications and Bases are that of the Improved Standard Technical Specifications for power reactors, to the extent they apply to a dry spent fuel storage cask system. NUMARC Document 93-03, "Writer's Guide for the Restructured Technical Specifications" [13.3.1] was used as a guide in the development of the Technical Specifications and Bases.

13.4 REGULATORY EVALUATION

Table 13.1.2 lists the Technical Specifications for the HI-STORM UMAX system. The Technical Specifications are detailed in Appendix A to the Certificate of Compliance. Authorized Contents (i.e., fuel specifications) and Design Features are provided in Appendix B to the CoC.

The conditions for use of the HI-STORM UMAX system identify necessary Technical Specifications, limits on authorized contents (i.e., fuel), and design features to satisfy 10 CFR Part 72, and the applicable acceptance criteria have been satisfied. Compliance with these Technical Specifications and other conditions of the Certificate of Compliance provides reasonable assurance that the HI-STORM UMAX system will provide safe storage of spent fuel and is in compliance with 10 CFR Part 72, the regulatory guides, applicable codes and standards, and accepted practices.

13.5 REFERENCES

[13.1.1] U.S. Code of Federal Regulations, Title 10, *Energy*, Part 72, *Licensing Requirements for Independent Storage of Spent Nuclear Fuel and High-Level Radioactive Waste*."

[13.1.2] U.S. Code of Federal Regulations, Title 10, *Energy*, Part 20, *Standards for Protection Against Radiation*."

[13.3.1] Nuclear Management and Resources Council, Inc. – *Writer's Guide for the Restructured Technical Specifications,* NUMARC 93-03, February 1993.

HI-STORM UMAX SYSTEM FSAR

APPENDIX 13.A

TECHNICAL SPECIFICATION BASES

FOR THE HOLTEC HI-STORM UMAX CANISTER STORAGE SYSTEM

BASES TABLE OF CONTENTS

Note: The text matter in the “arial” font is excerpted from the HI-STORM FW FSAR.

B 3.0 LIMITING CONDITION FOR OPERATION (LCO) APPLICABILITY

BASES

LCOs	LCO 3.0.1, 3.0.2, 3.0.4, and 3.0.5 establish the general requirements applicable to all Specifications and apply at all times, unless otherwise stated.
LCO 3.0.1	LCO 3.0.1 establishes the Applicability statement within each individual Specification as the requirement for when the LCO is required to be met (i.e., when the facility is in the specified conditions of the Applicability statement of each Specification).
LCO 3.0.2	LCO 3.0.2 establishes that upon discovery of a failure to meet an LCO, the associated ACTIONS shall be met. The Completion Time of each Required Action for an ACTIONS Condition is applicable from the point in time that an ACTIONS Condition is entered. The Required Actions establish those remedial measures that must be taken within specified Completion Times when the requirements of an LCO are not met. This Specification establishes that: a. Completion of the Required Actions within the specified Completion Times constitutes compliance with a Specification; and b. Completion of the Required Actions is not required when an LCO is met within the specified Completion Time, unless otherwise specified. There are two basic types of Required Actions. The first type of Required Action specifies a time limit in which the LCO must be met. This time limit is the Completion Time to restore a system or component or to restore variables to within specified limits. Whether stated as a Required Action or not, correction of the entered Condition is an action that may always be considered upon entering ACTIONS. The second type of Required Action specifies the remedial measures that permit continued operation that is not further restricted by the Completion Time. In this case, compliance with the Required Actions provides an acceptable level of safety for continued operation.

(continued)

BASES

LCO 3.0.2 (continued)	Completing the Required Actions is not required when an LCO is met or is no longer applicable, unless otherwise stated in the individual Specifications. The Completion Times of the Required Actions are also applicable when a system or component is removed from service intentionally. The reasons for intentionally relying on the ACTIONS include, but are not limited to, performance of Surveillances, preventive maintenance, corrective maintenance, or investigation of operational problems. Entering ACTIONS for these reasons must be done in a manner that does not compromise safety. Intentional entry into ACTIONS should not be made for operational convenience.
LCO 3.0.3	This specification is not applicable to a dry storage cask system because it describes conditions under which a power reactor must be shut down when an LCO is not met and an associated ACTION is not met or provided. The placeholder is retained for consistency with the power reactor technical specifications.
LCO 3.0.4	LCO 3.0.4 establishes limitations on changes in specified conditions in the Applicability when an LCO is not met. It precludes placing the HI-STORM UMAX System in a specified condition stated in that Applicability (e.g., Applicability desired to be entered) when the following exist: a. Facility conditions are such that the requirements of the LCO would not be met in the Applicability desired to be entered; and b. Continued noncompliance with the LCO requirements, if the Applicability were entered, would result in being required to exit the Applicability desired to be entered to comply with the Required Actions. Compliance with Required Actions that permit continuing with dry fuel storage activities for an unlimited period of time in a specified condition provides an acceptable level of safety for continued operation. This is without regard to the status of the dry storage system. Therefore, in such cases, entry into a specified condition in the Applicability may be made in accordance with the provisions of the Required Actions. The provisions of this Specification should not be interpreted as endorsing the failure to exercise the good practice of restoring systems or components before entering an associated specified condition in the Applicability.

(continued)

BASES	
LCO 3.0.4 (continued)	The provisions of LCO 3.0.4 shall not prevent changes in specified conditions in the Applicability that are required to comply with ACTIONS. In addition, the provisions of LCO 3.0.4 shall not prevent changes in specified conditions in the Applicability that are related to the unloading of an SFSC. Exceptions to LCO 3.0.4 are stated in the individual Specifications. Exceptions may apply to all the ACTIONS or to a specific Required Action of a Specification.
LCO 3.0.5	LCO 3.0.5 establishes the allowance for restoring equipment to service under administrative controls when it has been removed from service or determined to not meet the LCO to comply with the ACTIONS. The sole purpose of this Specification is to provide an exception to LCO 3.0.2 (e.g., to not comply with the applicable Required Action(s)) to allow the performance of testing to demonstrate: a. The equipment being returned to service meets the LCO; or b. Other equipment meets the applicable LCOs. The administrative controls ensure the time the equipment is returned to service in conflict with the requirements of the ACTIONS is limited to the time absolutely necessary to perform the allowed testing. This Specification does not provide time to perform any other preventive or corrective maintenance.

B 3.0 SURVEILLANCE REQUIREMENT (SR) APPLICABILITY

BASES

SRs	SR 3.0.1 through SR 3.0.4 establish the general requirements applicable to all Specifications and apply at all times, unless otherwise stated.
SR 3.0.1	SR 3.0.1 establishes the requirement that SRs must be met during the specified conditions in the Applicability for which the requirements of the LCO apply, unless otherwise specified in the individual SRs. This Specification is to ensure that Surveillances are performed to verify that systems and components meet the LCO and variables are within specified limits. Failure to meet a Surveillance within the specified Frequency, in accordance with SR 3.0.2, constitutes a failure to meet an LCO.

Systems and components are assumed to meet the LCO when the associated SRs have been met. Nothing in this Specification, however, is to be construed as implying that systems or components meet the associated LCO when:

a. The systems or components are known to not meet the LCO, although still meeting the SRs; or

b. The requirements of the Surveillance(s) are known to be not met between required Surveillance performances.

Surveillances do not have to be performed when the HI-STORM UMAX System is in a specified condition for which the requirements of the associated LCO are not applicable, unless otherwise specified.

Surveillances, including Surveillances invoked by Required Actions, do not have to be performed on equipment that has been determined to not meet the LCO because the ACTIONS define the remedial measures that apply. Surveillances have to be met and performed in accordance with SR 3.0.2, prior to returning equipment to service. Upon completion of maintenance, appropriate post-maintenance testing is required. This includes ensuring applicable Surveillances are not failed and their most recent performance is in accordance with SR 3.0.2. Post maintenance testing may not be possible in the current specified conditions in the Applicability due to the necessary dry storage cask system parameters not having been established. In these situations, the equipment may be considered to meet the LCO provided testing has been satisfactorily completed to the extent possible and the equipment is not otherwise believed to be incapable of performing its function. This will allow dry fuel storage activities to proceed to a specified condition where other necessary post maintenance tests can be completed.

(continued)

BASES

SR 3.0.2

SR 3.0.2 establishes the requirements for meeting the specified Frequency for Surveillances and any Required Action with a Completion Time that requires the periodic performance of the Required Action on a "once per..." interval.

SR 3.0.2 permits a 25% extension of the interval specified in the Frequency. This extension facilitates Surveillance scheduling and considers facility conditions that may not be suitable for conducting the Surveillance (e.g., transient conditions or other ongoing Surveillance or maintenance activities).

The 25% extension does not significantly degrade the reliability that results from performing the Surveillance at its specified Frequency. This is based on the recognition that the most probable result of any particular Surveillance being performed is the verification of conformance with the SRs. The exceptions to SR 3.0.2 are those Surveillances for which the 25% extension of the interval specified in the Frequency does not apply. These exceptions are stated in the individual Specifications as a Note in the Frequency stating, "SR 3.0.2 is not applicable."

As stated in SR 3.0.2, the 25% extension also does not apply to the initial portion of a periodic Completion Time that requires performance on a "once per..." basis. The 25% extension applies to each performance after the initial performance. The initial performance of the Required Action, whether it is a particular Surveillance or some other remedial action, is considered a single action with a single Completion Time. One reason for not allowing the 25% extension to this Completion Time is that such an action usually verifies that no loss of function has occurred by checking the status of redundant or diverse components or accomplishes the function of the affected equipment in an alternative manner.
The provisions of SR 3.0.2 are not intended to be used repeatedly merely as an operational convenience to extend Surveillance intervals or periodic Completion Time intervals beyond those specified.

(continued)

BASES

SR 3.0.3 SR 3.0.3 establishes the flexibility to defer declaring affected equipment as not meeting the LCO or an affected variable outside the specified limits when a Surveillance has not been completed within the specified Frequency. A delay period of up to 24 hours or up to the limit of the specified Frequency, whichever is less, applies from the point in time that it is discovered that the Surveillance has not been performed in accordance with SR 3.0.2, and not at the time that the specified Frequency was not met.

This delay period provides adequate time to complete Surveillances that have been missed. This delay period permits the completion of a Surveillance before complying with Required Actions or other remedial measures that might preclude completion of the Surveillance.

The basis for this delay period includes consideration of HI-STORM UMAX System conditions, adequate planning, availability of personnel, the time required to perform the Surveillance, the safety significance of the delay in completing the required Surveillance, and the recognition that the most probable result of any particular Surveillance being performed is the verification of conformance with the requirements. When a Surveillance with a Frequency based not on time intervals, but upon specified facility conditions, is discovered not to have been performed when specified, SR 3.0.3 allows the full delay period of 24 hours to perform the Surveillance.

SR 3.0.3 also provides a time limit for completion of Surveillances that become applicable as a consequence of changes in the specified conditions in the Applicability imposed by the Required Actions.

Failure to comply with specified Frequencies for SRs is expected to be an infrequent occurrence. Use of the delay period established by SR 3.0.3 is a flexibility which is not intended to be used as an operational convenience to extend Surveillance intervals.

(continued)

BASES

SR 3.0.3 (continued)	If a Surveillance is not completed within the allowed delay period, then the equipment is considered to not meet the LCO or the variable is considered outside the specified limits and the Completion Times of the Required Actions for the applicable LCO Conditions begin immediately upon expiration of the delay period. If a Surveillance is failed within the delay period, then the equipment does not meet the LCO, or the variable is outside the specified limits and the Completion Times of the Required Actions for the applicable LCO Conditions begin immediately upon the failure of the Surveillance. Completion of the Surveillance within the delay period allowed by this Specification, or within the Completion Time of the ACTIONS, restores compliance with SR 3.0.1.
SR 3.0.4	SR 3.0.4 establishes the requirement that all applicable SRs must be met before entry into a specified condition in the Applicability. This Specification ensures that system and component requirements and variable limits are met before entry into specified conditions in the Applicability for which these systems and components ensure safe conduct of dry fuel storage activities. The provisions of this Specification should not be interpreted as endorsing the failure to exercise the good practice of restoring systems or components before entering an associated specified condition in the Applicability. However, in certain circumstances, failing to meet an SR will not result in SR 3.0.4 restricting a change in specified condition. When a system, subsystem, division, component, device, or variable is outside its specified limits, the associated SR(s) are not required to be performed per SR 3.0.1, which states that Surveillances do not have to be performed on equipment that has been determined to not meet the LCO. When equipment does not meet the LCO, SR 3.0.4 does not apply to the associated SR(s) since the requirement for the SR(s) to be performed is removed. Therefore, failing to perform the Surveillance(s) within the specified Frequency does not result in an SR 3.0.4 restriction to changing specified conditions of the Applicability. However, since the LCO is not met in this instance, LCO 3.0.4 will govern any restrictions that may (or may not) apply to specified condition changes.

(continued)

BASES	
SR 3.0.4 (continued)	The provisions of SR 3.0.4 shall not prevent changes in specified conditions in the Applicability that are required to comply with ACTIONS. In addition, the provisions of LCO 3.0.4 shall not prevent changes in specified conditions in the Applicability that are related to the unloading of an SFSC. The precise requirements for performance of SRs are specified such that exceptions to SR 3.0.4 are not necessary. The specific time frames and conditions necessary for meeting the SRs are specified in the Frequency, in the Surveillance, or both. This allows performance of Surveillances when the prerequisite condition(s) specified in a Surveillance procedure require entry into the specified condition in the Applicability of the associated LCO prior to the performance or completion of a Surveillance. A Surveillance that could not be performed until after entering the LCO Applicability would have its Frequency specified such that it is not "due" until the specific conditions needed are met. Alternately, the Surveillance may be stated in the form of a Note as not required (to be met or performed) until a particular event, condition, or time has been reached. Further discussion of the specific formats of SRs' annotation is found in Section 1.4, Frequency.

B 3.1 SFSC Integrity

B 3.1.1 Multi-Purpose Canister (MPC)

BASES

BACKGROUND

A TRANSFER CASK with an empty MPC is placed in the spent fuel pool and loaded with fuel assemblies meeting the requirements of the CoC. A lid is then placed on the MPC. The TRANSFER CASK and MPC are raised to the top of the spent fuel pool surface. The TRANSFER CASK and MPC are then moved into the preparation area where the MPC lid is welded to the MPC shell and the welds are inspected and tested. The water is drained from the MPC cavity and drying is performed. The MPC cavity is backfilled with helium. Then, the MPC vent and drain port cover plates and closure ring are installed and welded. Inspections are performed on the welds.

MPC cavity moisture removal using vacuum drying or forced helium dehydration is performed to remove residual moisture from the MPC cavity space after the MPC has been drained of water. If vacuum drying is used, any water that has not drained from the fuel cavity evaporates from the fuel cavity due to the vacuum. This is aided by the temperature increase due to the decay heat of the fuel.

If forced helium dehydration is used, the dry gas introduced to the MPC cavity through the vent or drain port absorbs the residual moisture in the MPC. This humidified gas exits the MPC via the other port and the absorbed water is removed through condensation and/or mechanical drying. The dried helium is then forced back to the MPC until the temperature acceptance limit is met.

After the completion of drying, the MPC cavity is backfilled with helium meeting the requirements of the CoC.

Backfilling of the MPC fuel cavity with helium promotes gaseous heat dissipation and the inert atmosphere protects the fuel cladding. Backfilling the MPC with helium in the required quantity eliminates air in-leakage over the life of the MPC because the cavity pressure rises due to heat up of the confined gas by the fuel decay heat during storage.

(continued)

BASES	
APPLICABLE SAFETY ANALYSIS	The confinement of radioactivity during the storage of spent fuel in the MPC is ensured by the confinement boundary of the MPC in which the fuel assemblies are stored. Long-term integrity of the fuel and cladding depend on storage in an inert atmosphere. This is accomplished by removing water from the MPC and backfilling the cavity with an inert gas. The thermal analyses of the MPC assume that the MPC cavity is filled with dry helium of a minimum quantity to ensure the assumptions used for convection heat transfer are preserved. Keeping the backfill pressure below the maximum value preserves the initial condition assumptions made in the MPC over-pressurization evaluation.
LCO	A dry, helium filled, and sealed MPC establishes an inert heat removal environment necessary to ensure the integrity of the fuel cladding. Moreover, it also ensures that there will be no air in-leakage into the MPC cavity that could damage the fuel cladding over the storage period.
APPLICABILITY	The dry, sealed, and inert atmosphere is required to be in place prior to TRANSPORT OPERATIONS to ensure both the confinement and heat removal mechanisms are in place during these operating periods. These conditions are not required during LOADING OPERATIONS or UNLOADING OPERATIONS as these conditions are being established or removed, respectively, during these periods in support of other activities being performed with the stored fuel.

(continued)

BASES

ACTIONS

A note has been added to the ACTIONS which states that, for this LCO, separate Condition entry is allowed for each MPC. This is acceptable since the Required Actions for each Condition provide appropriate compensatory measures for each MPC not meeting the LCO. Subsequent MPCs that do not meet the LCO are governed by subsequent Condition entry and application of associated Required Actions.

A.1

If the cavity vacuum drying pressure or demoisturizer exit gas temperature limit has been determined not to be met prior to TRANSPORT OPERATIONS, an engineering evaluation is necessary to determine the potential quantity of moisture left within the MPC cavity. Since moisture remaining in the cavity during these modes of operation represent a long-term degradation concern, immediate action is not necessary. The Completion Time is sufficient to complete the engineering evaluation commensurate with the safety significance of the CONDITION.

A.2

Once the quantity of moisture potentially left in the MPC cavity is determined, a corrective action plan shall be developed and actions initiated to the extent necessary to return the MPC to an analyzed condition. Since the quantity of moisture estimated under Required Action A.1 can range over a broad scale, different recovery strategies may be necessary. Since moisture remaining in the cavity during these modes of operation represent a long-term degradation concern, immediate action is not necessary. The Completion Time is sufficient to develop and initiate the corrective actions commensurate with the safety significance of the CONDITION.

(continued)

BASES	
ACTIONS (continued)	B.1 If the helium backfill quantity limit has been determined not to be met prior to TRANSPORT OPERATIONS, an engineering evaluation is necessary to determine the quantity of helium within the MPC cavity. Since too much or too little helium in the MPC during these modes represents a potential overpressure or heat removal degradation concern, an engineering evaluation shall be performed in a timely manner. The Completion Time is sufficient to complete the engineering evaluation commensurate with the safety significance of the CONDITION. B.2 Once the quantity of helium in the MPC cavity is determined, a corrective action plan shall be developed and initiated to the extent necessary to return the MPC to an analyzed condition either by adding or removing helium or by demonstrating through analysis that all system limits will continue to be met. Since the quantity of helium estimated under Required Action B.1 can range over a broad scale, different recovery strategies may be necessary. Since elevated or reduced helium quantities existing in the MPC cavity represent a potential overpressure or heat removal degradation concern, corrective actions should be developed and implemented in a timely manner. The Completion Time is sufficient to develop and initiate the corrective actions commensurate with the safety significance of the CONDITION. C.1 If the helium leak rate limit has been determined not to be met prior to TRANSPORT OPERATIONS, an engineering evaluation is necessary to determine the impact of increased helium leak rate on heat removal and off-site dose. Since the HI-STORM UMAX VVM is a ventilated system, any leakage from the MPC is transported directly to the environment. Since an increased helium leak rate represents a potential challenge to MPC heat removal and the off-site doses, reasonably rapid action is warranted. The Completion Time is sufficient to complete the engineering evaluation commensurate with the safety significance of the CONDITION.

(continued)

BASES	
ACTIONS (continued)	C.2 Once the consequences of the elevated leak rate from the MPC are determined, a corrective action plan shall be developed and initiated to the extent necessary to return the MPC to an analyzed condition. Since the recovery mechanisms can range over a broad scale based on the evaluation performed under Required Action C.1, different recovery strategies may be necessary. Since an elevated helium leak rate represents a challenge to heat removal rates and offsite doses, reasonably rapid action is required. The Completion Time is sufficient to develop and initiate the corrective actions commensurate with the safety significance of the CONDITION. D.1 If the MPC fuel cavity cannot be successfully returned to a safe, analyzed condition, the fuel must be placed in a safe condition in the spent fuel pool. The Completion Time is reasonable based on the time required to re-flood the MPC, cut the MPC lid welds, move the TRANSFER CASK into the spent fuel pool, remove the MPC lid, and remove the spent fuel assemblies in an orderly manner and without challenging personnel.
SURVEILLANCE REQUIREMENTS	SR 3.1.1.1 , SR 3.1.1.2, and SR 3.1.1.3 The long-term integrity of the stored fuel is dependent on storage in a dry, inert environment. Under certain conditions, cavity dryness may be demonstrated either by evacuating the cavity to a very low absolute pressure and verifying that the pressure is held over a specified period of time or by recirculating dry helium through the MPC cavity to absorb moisture until the gas temperature or dew point at the specified location reaches and remains below the acceptance limit for the specified time period. A low vacuum pressure or a demoisturizer exit temperature meeting the acceptance limit is an indication that the cavity is dry. Other conditions require the forced helium dehydration method of moisture removal to be used to provide necessary cooling of the fuel during drying operations.

(continued)

BASES	
SURVEILLANCE REQUIREMENTS (continued)	Cooling provided by normal operation of the forced helium dehydration system ensures that the fuel cladding temperature remains below the applicable limits since forced recirculation of helium provides more effective heat transfer than that which occurs during normal storage operations. The conditions and requirements for drying the MPC cavity based on the burnup class of the fuel (moderate or high), heat load, and the applicable short-term temperature limit are given in the CoC/TS Appendix A, Table 3-1. The temperature limits and associated cladding hoop stress calculation requirements are consistent with the guidance in NRC Interim Staff Guidance (ISG) Document 11. Having the proper quantity of helium in the MPC ensures adequate heat transfer from the fuel to the fuel basket and surrounding structure of the MPC and precludes any overpressure event from challenging the normal, off-normal, or accident design pressure of the MPC. Meeting the helium leak rate limit prior to TRANSPORT OPERATIONS ensures there is adequate helium in the MPC for long term storage and that there is no credible effluent dose from the MPC. All of these surveillances must be successfully performed once, prior to TRANSPORT OPERATIONS to ensure that the conditions are established for SFSC storage which preserve the analysis basis supporting the MPC design.
REFERENCES	1. FSAR Chapters 1, 4, 7 and 9 of the HI-STORM UMAX and HI-STORM FW FSARs 2. Interim Staff Guidance Document 11, Rev. 3 3. Interim Staff Guidance Document 18, Rev. 1

B 3.1 SFSC Integrity

B 3.1.2 SFSC Heat Removal System

BASES	
BACKGROUND	The SFSC Heat Removal System is a passive, air-cooled, convective heat transfer system that ensures heat from the MPC canister is transferred to the environs by the chimney effect. Air is drawn into the inlet ducts and travels down the space between the Cavity Enclosure Container (CEC) and the Divider Shell, through the cut-outs at the bottom of the Divider Shell, up the space between the Divider Shell and the MPC, and out through the outlet duct. The MPC transfers its heat from its surface to the air via natural convection. The buoyancy created by the heating of the air creates a chimney effect.
APPLICABLE SAFETY ANALYSIS	The thermal analyses of the SFSC take credit for the decay heat from the spent fuel assemblies being ultimately transferred to the ambient environment surrounding the VVM. Transfer of heat away from the fuel assemblies ensures that the fuel cladding and other SFSC component temperatures do not exceed applicable limits. Under normal storage conditions, the inlet and outlet duct screens are unobstructed and full air flow occurs. Analyses have been performed for half and complete obstruction of the inlet duct screens. Blockage of half of the inlet ducts reduces air flow through the VVM and decreases heat transfer from the MPC. Under this off-normal condition, no SFSC components exceed the short term temperature limits. The complete blockage of all inlet air ducts stops normal air cooling of the MPC. The MPC will continue to radiate heat to the relatively cooler subgrade. With the loss of normal air cooling, the SFSC component temperatures will increase toward their respective short-term temperature limits. None of the components reach their temperature limits over the duration of the analyzed event.

(continued)

BASES	
LCO	The SFSC Heat Removal System must be verified to be operable to preserve the assumptions of the thermal analyses. Operability is defined as 50% or more of the inlet vent duct areas are unblocked and available for flow. Operability of the heat removal system ensures that the decay heat generated by the stored fuel assemblies is transferred to the environs at a sufficient rate to maintain fuel cladding and other SFSC component temperatures within design limits. The intent of this LCO is to address those occurrences of air duct screen blockage that can be reasonably anticipated to occur from time to time at the ISFSI (i.e., Design Event I and II class events per ANSI/ANS-57.9). These events are of the type where corrective actions can usually be accomplished within one 8-hour operating shift to restore the heat removal system to operable status (e.g., removal of loose debris). This LCO is not intended to address low frequency, unexpected Design Event III and IV class events (ANSI/ANS-57.9) such as design basis accidents and extreme environmental phenomena that could potentially block one or more of the air ducts for an extended period of time (i.e., longer than the total Completion Time of the LCO). This class of events is addressed site-specifically as required by Section 3.4.11 of Appendix B to the CoC.
APPLICABILITY	The LCO is applicable during STORAGE OPERATIONS. Once a VVM containing an MPC loaded with spent fuel has been placed in storage, the heat removal system must be operable to ensure adequate dissipation of the decay heat from the fuel assemblies.
ACTIONS	A note has been added to the ACTIONS which states that, for this LCO, separate Condition entry is allowed for each SFSC. This is acceptable since the Required Actions for each Condition provide appropriate compensatory measures for each SFSC not meeting the LCO. Subsequent SFSCs that don't meet the LCO are governed by subsequent Condition entry and application of associated Required Actions.

(continued)

BASES

ACTIONS (continued)

A.1

Although the heat removal system remains operable, the blockage should be cleared expeditiously.

B.1

If the heat removal system has been determined to be inoperable, it must be restored to operable status within eight hours. Eight hours is a reasonable period of time to take action to remove the obstructions in the air flow path.

C.1

If the heat removal system cannot be restored to operable status within eight hours, the VVM and the fuel may experience elevated temperatures. Therefore, dose rates are required to be measured to verify the effectiveness of the radiation shielding provided by the concrete. This Action must be performed immediately and repeated every twelve hours thereafter to provide timely and continued evaluation of the effectiveness of the concrete shielding. As necessary, the system user shall provide additional radiation protection measures such as temporary shielding. The Completion Time is reasonable considering the expected slow rate of deterioration, if any, of the concrete under elevated temperatures.

C.2.1

In addition to Required Action C.1, efforts must continue to restore cooling to the SFSC. Efforts must continue to restore the heat removal system to operable status by removing the air flow obstruction(s) unless optional Required Action C.2.2 is being implemented.

This Required Action must be complete in 24 hours. The Completion Time is consistent with the thermal analyses of this event, which show that all component temperatures remain below their short-term temperature limits up to 32 hours after event initiation.

(continued)

BASES

ACTIONS (continued)

C.2.1 (continued)

The Completion Time reflects the 8 hours to complete Required Action B.1 and the appropriate balance of time consistent with the applicable analysis results. The event is assumed to begin at the time the SFSC heat removal system is declared inoperable. This is reasonable considering the low probability of all inlet ducts becoming simultaneously blocked.

C.2.2

In lieu of implementing Required Action C.2.1, transfer of the MPC into a TRANSFER CASK will place the MPC in an analyzed condition and ensure adequate fuel cooling until actions to correct the heat removal system inoperability can be completed. Transfer of the MPC into a TRANSFER CASK removes the SFSC from the LCO Applicability since STORAGE OPERATIONS does not include times when the MPC resides in the TRANSFER CASK.

An engineering evaluation must be performed to determine if any deterioration which prevents the VVM from performing its design function. If the evaluation is successful and the air inlet duct screens have been cleared, the VVM heat removal system may be considered operable and the MPC transferred back into the VVM. Compliance with LCO 3.1.2 is then restored. If the evaluation is unsuccessful, the user must transfer the MPC into a different, fully qualified VVM to resume STORAGE OPERATIONS and restore compliance with LCO 3.1.2

In lieu of performing the engineering evaluation, the user may opt to proceed directly to transferring the MPC into a different, fully qualified VVM or place the TRANSFER CASK in the spent fuel pool and unload the MPC.

The Completion Time of 24 hours reflects the Completion Time from Required Action C.2.1 to ensure component temperatures remain below their short-term temperature limits for the respective decay heat loads.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.2

The long-term integrity of the stored fuel is dependent on the ability of the SFSC to reject heat from the MPC to the environment. There are two options for implementing SR 3.1.2, either of which is acceptable for demonstrating that the heat removal system is OPERABLE.

Visual observation that all air inlet duct screens are unobstructed ensures that the SFSC is operable. If greater than 50% of the air inlet duct screens are blocked the heat removal system is inoperable and this LCO is not met. While 50% or less blockage of the total air inlet duct screen area does not constitute inoperability of the heat removal system, corrective actions should be taken promptly to remove the obstruction and restore full flow.

Visual observation of air outlet duct screen blockage does not constitute inoperability of the heat removal system; however, corrective action should be taken to promptly remove the obstruction.

As an alternative, for VVMs with air temperature monitoring instrumentation installed in the air outlets, the temperature difference between the outlet air and the ambient air may be monitored to verify operability of the heat removal system. Blocked air inlet duct screens will reduce air flow and increase the outlet duct air temperature. Based on the analyses, if the temperature difference between the ambient air and the outlet duct air meets the criteria in the LCO, adequate air flow is occurring to provide assurance of long term fuel cladding integrity. The reference ambient temperature used to perform this Surveillance shall be measured at the ISFSI facility.

The Frequency of 24 hours is reasonable based on the time necessary for SFSC components to heat up to unacceptable temperatures assuming design basis heat loads, and allowing for corrective actions to take place upon discovery of blockage of air ducts.

REFERENCES

1. FSAR Chapter 4
2. ANSI/ANS 57.9-1992

B 3.1 SFSC INTEGRITY

B 3.1.3 MPC Cavity Reflooding

BASES	
BACKGROUND	In the event that an MPC must be unloaded, the TRANSFER CASK with its enclosed MPC is returned to the preparation area to begin the process of fuel unloading. The MPC closure ring, and vent and drain port cover plates are removed. The MPC gas is sampled to determine the integrity of the spent fuel cladding. The pressure in the MPC cavity is ensured to be less than the 100 psig design pressure. This is accomplished via direct measurement of the MPC gas pressure or via analysis. After ensuring the MPC cavity pressure meets the LCO limit, the MPC is then reflooded with water at a controlled rate and/or the pressure monitored to ensure that the pressure remains below 100 psig. Once the cavity is filled with water, the MPC lid weld is removed leaving the MPC lid in place. The TRANSFER CASK and MPC are placed in the spent fuel pool and the MPC lid is removed. The fuel assemblies are removed from the MPC and the MPC and TRANSFER CASK are removed from the spent fuel pool and decontaminated. Ensuring that the MPC cavity pressure is less than the LCO limit ensures that any steam produced within the cavity is safely vented to an appropriate location and eliminates the risk of high MPC pressure due to sudden generation of large steam quantities during re-flooding.
APPLICABLE SAFETY ANALYSIS	The confinement of radioactivity during the storage of spent fuel in the MPC is ensured by the MPC in which the fuel assemblies are stored. Standard practice in the dry storage industry has historically been to directly reflood the storage canister with water. This standard practice is known not to induce fuel cladding failures. The integrity of the MPC depends on maintaining the internal cavity pressures within design limits. This is accomplished by introducing water to the cavity in a controlled manner such that there is no sudden formation of large quantities of steam during MPC reflooding. (Ref. 1).

(continued)

BASES	
LCO	Determining the MPC cavity pressure prior to and during re-flooding ensures that there will be sufficient venting of any steam produced to avoid excessive MPC pressurization.
APPLICABILITY	The MPC cavity pressure is controlled during UNLOADING OPERATIONS after the TRANSFER CASK and integral MPC are back in the FUEL BUILDING and are no longer suspended from, or secured in, the transporter. Therefore, the MPC Reflood LCO does not apply during TRANSPORT OPERATIONS and STORAGE OPERATIONS. A note has been added to the APPLICABILITY for LCO 3.1.3 which states that the LCO is only applicable during wet UNLOADING OPERATIONS. This is acceptable since the intent of the LCO is to avoid uncontrolled MPC pressurization due to water flashing during re-flooding operations. This is not a concerning for dry UNLOADING OPERATIONS.
ACTIONS	A note has been added to the ACTIONS which states that, for this LCO, separate Condition entry is allowed for each MPC. This is acceptable since the Required Actions for each Condition provide appropriate compensatory measures for each MPC not meeting the LCO. Subsequent MPCs that do not meet the LCO are governed by subsequent Condition entry and application of associated Required Actions. A If the MPC cavity pressure limit is not met, actions must be taken to restore the parameters to within the limits before initiating or continuing re-flooding the MPC. Immediately is an appropriate Completion Time because it requires action to be initiated promptly and completed without delay, but does not establish any particular fixed time limit for completing the action. This offers the flexibility necessary for users to plan and implement any necessary work activities commensurate with the safety significance of the condition, which is governed by the MPC heat load.

(continued)

BASES	
SURVEILLANCE REQUIREMENTS	SR 3.1.3.1 The integrity of the MPC is dependent on controlling the internal MPC pressure. By controlling the MPC internal pressure prior to and during re-flooding the MPC, sufficient steam venting capacity exists during MPC re-flooding. The LCO must be met on each SFSC before the initiation of MPC re-flooding operations to ensure the design and analysis basis are preserved.
REFERENCES	1. FSAR Chapters 3, 4, 9 and 12 of HI-STORM FW FSAR

B 3.2 SFSC Radiation Protection

B 3.2.1 TRANSFER CASK Surface Contamination

BASES	
BACKGROUND	A TRANSFER CASK is immersed in the spent fuel pool in order to load the spent fuel assemblies. As a result, the surface of the TRANSFER CASK may become contaminated with the radioactive material from the spent fuel pool water. This contamination is removed prior to moving the TRANSFER CASK to the ISFSI, in order to minimize the radioactive contamination to personnel or the environment. This allows dry fuel storage activities to proceed without additional radiological controls to prevent the spread of contamination and reduces personnel dose due to the spread of loose contamination or airborne contamination. This is consistent with ALARA practices.
APPLICABLE SAFETY ANALYSIS	The radiation protection measures, implemented during MPC transfer and transportation using the TRANSFER CASK, are based on the assumption that the exterior surfaces of the TRANSFER CASK have been decontaminated. Failure to decontaminate the surfaces of the TRANSFER CASK could lead to higher-than-projected occupational doses.
LCO	Removable surface contamination on the TRANSFER CASK exterior surfaces and accessible surfaces of the MPC is limited to 1000 dpm/100 cm^2 from beta and gamma sources and 20 dpm/100 cm^2 from alpha sources. These limits are taken from the guidance in IE Circular 81-07 (Ref. 2) and are based on the minimum level of activity that can be routinely detected under a surface contamination control program using direct survey methods. Only loose contamination is controlled, as fixed contamination will not result from the TRANSFER CASK loading process.

(continued)

BASES

LCO (continued)

Experience has shown that these limits are low enough to prevent the spread of contamination to clean areas and are significantly less than the levels which would cause significant personnel skin dose. LCO 3.2.1 requires removable contamination to be within the specified limits for the exterior surfaces of the TRANSFER CASK and accessible portions of the MPC. The location and number of surface swipes used to determine compliance with this LCO are determined based on standard industry practice and the user's plant-specific contamination measurement program for objects of this size. Accessible portions of the MPC means the upper portion of the MPC external shell wall accessible after the inflatable annulus seal is removed and before the annulus shield ring is installed. The user shall determine a reasonable number and location of swipes for the accessible portion of the MPC. The objective is to determine a removable contamination value representative of the entire upper circumference of the MPC, while implementing sound ALARA practices.

APPLICABILITY

Verification that the surface contamination is less than the limit in the LCO is performed during LOADING OPERATIONS. This occurs before TRANSPORT OPERATIONS, when the LCO is applicable. Measurement of surface contamination is unnecessary during UNLOADING OPERATIONS as surface contamination would have been measured prior to moving the subject TRANSFER CASK to the ISFSI.

(continued)

BASES	
ACTIONS	A note has been added to the ACTIONS which states that, for this LCO, separate Condition entry is allowed for each TRANSFER CASK. This is acceptable since the Required Actions for each Condition provide appropriate compensatory measures for each TRANSFER CASK not meeting the LCO. A subsequent use of the TRANSFER CASK that does not meet the LCO are governed by subsequent Condition entry and application of associated Required Actions. <u>A.1</u> If the removable surface contamination of a TRANSFER CASK or MPC, as applicable, which has been loaded with spent fuel is not within the LCO limits, action must be initiated to decontaminate the TRANSFER CASK or MPC and bring the removable surface contamination to within limits. The Completion Time of 7 days is appropriate given that sufficient time is needed to prepare for, and complete the decontamination once the LCO is determined not to be met.
SURVEILLANCE REQUIREMENTS	SR 3.2.1.1 This SR verifies that the removable surface contamination on the TRANSFER CASK and/or accessible portions of the MPC is less than the limits in the LCO. The Surveillance is performed using smear surveys to detect removable surface contamination. The Frequency requires performing the verification during LOADING OPERATIONS in order to confirm that the TRANSFER CASK or VVM can be moved to the ISFSI without spreading loose contamination.
REFERENCES	1. HI-STORM FW FSAR Chapter 9 2. NRC IE Circular 81-07.

B 3.3 SFSC Criticality Control

B 3.3.1 Boron Concentration

BASES	
BACKGROUND	A TRANSFER CASK with an empty MPC is placed in the spent fuel pool and loaded with fuel assemblies meeting the requirements of the Certificate of Compliance. A lid is then placed on the MPC. The TRANSFER CASK and MPC are raised to the top of the spent fuel pool surface. The TRANSFER CASK and MPC are then moved into the preparation area where the MPC lid is welded to the MPC shell and the welds are inspected and tested. The water is drained from the MPC cavity and drying is performed. The MPC cavity is backfilled with helium. Then, the MPC vent and drain cover plates and MPC closure ring are installed and welded. Inspections are performed on the welds. For those MPCs containing PWR fuel assemblies credit is taken in the criticality analyses for boron in the water within the MPC. To preserve the analysis basis, users must verify that the boron concentration of the water in the MPC meets specified limits when there is fuel and water in the MPC. This may occur during LOADING OPERATIONS and UNLOADING OPERATIONS.
APPLICABLE SAFETY ANALYSIS	The spent nuclear fuel stored in the SFSC is required to remain subcritical ($k_{eff} \leq 0.95$) under all conditions of storage. The HI-STORM UMAX SFSC is analyzed to store a wide variety of spent nuclear fuel assembly types with differing initial enrichments. For all PWR fuel loaded in the MPC-37, credit was taken in the criticality analyses for neutron poison in the form of soluble boron in the water within the MPC. Compliance with this LCO preserves the assumptions made in the criticality analyses regarding credit for soluble boron.

(continued)

BASES	
LCO	Compliance with this LCO ensures that the stored fuel will remain subcritical with a $k_{eff} \leq 0.95$ while water is in the MPC. LCOs 3.3.1.a provides the minimum concentration of soluble boron required in the MPC water for the MPC-37. The amount of soluble boron is dependent on the initial enrichment of the fuel assemblies to be loaded in the MPC. Fuel assemblies with an initial enrichment less than or equal to 4.0 wt. % U-235 require less soluble boron than those with initial enrichments greater than 4.0 wt. % U-235. For initial enrichments greater than 4.0 wt. % U-235 and up to 5.0 wt. % U-225, interpolation is permitted to determine the required minimum amount of soluble boron. All fuel assemblies loaded into the MPC-37 are limited by analysis to maximum enrichments of 5.0 wt. % U-235. The LCO also requires that the minimum soluble boron concentration for the most limiting fuel assembly array/class and classification to be stored in the same MPC be used. This means that the highest minimum soluble boron concentration limit for all fuel assemblies in the MPC applies in cases where fuel assembly array/classes are mixed in the same MPC. This ensures the assumptions pertaining to soluble boron used in the criticality analyses are preserved.
APPLICABILITY	The boron concentration LCO is applicable whenever an MPC-37 has at least one PWR fuel assembly in a storage location and water in the MPC. During LOADING OPERATIONS, the LCO is applicable immediately upon the loading of the first fuel assembly in the MPC. It remains applicable until the MPC is drained of water. During UNLOADING OPERATIONS, the LCO is applicable when the MPC is reflooded with water. Note that compliance with SR 3.0.4 assures that the water to be used to flood the MPC is of the correct boron concentration to ensure the LCO is satisfied upon entering the Applicability.

(continued)

BASES

ACTIONS

A note has been added to the ACTIONS which states that, for this LCO, separate Condition entry is allowed for each MPC. This is acceptable since the Required Actions for each Condition provide appropriate compensatory measures for each MPC not meeting the LCO. Subsequent MPCs that do not meet the LCO are governed by subsequent Condition entry and application of associated Required Actions.

A.1 and A.2

Continuation of LOADING OPERATIONS, UNLOADING OPERATIONS or positive reactivity additions (including actions to reduce boron concentration) is contingent upon maintaining the SFSC in compliance with the LCO. If the boron concentration of water in the MPC is less than its limit, all LOADING OPERATIONS, UNLOADING OPERATIONS or positive reactivity additions must be suspended immediately.

A.3

In addition to immediately suspending LOADING OPERATIONS, UNLOADING OPERATIONS and positive reactivity additions, action to restore the concentration to within the limit specified in the LCO must be initiated immediately. One means of complying with this action is to initiate boration of the affected MPC. In determining the required combination of boration flow rate and concentration, there is no unique design basis event that must be satisfied; only that boration is initiated without delay. In order to raise the boron concentration as quickly as possible, the operator should begin boration with the best source available for existing plant conditions.

Once boration is initiated, it must be continued until the boron concentration is restored. The restoration time depends on the amount of boron that must be injected to reach the required concentration.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.1

The boron concentration in the MPC water must be verified to be within the applicable limit within four hours prior to entering the Applicability of the LCO. For LOADING OPERATIONS, this means within four hours of loading the first fuel assembly into the MPC using two independent measurements to ensure the requirements of 10 CFR 72.124(a) are met. These two independent measurements will be repeated every 48 hours while the MPC is submerged in water or if water is to be added to or recirculated through the MPC.

For UNLOADING OPERATIONS, this means verifying the boron concentration in the source of borated water to be used to reflood the MPC within four hours of commencing reflooding operations and every 48 hours after until all the fuel is removed from the MPC. Two independent measurements will be taken to ensure the requirements of 10 CFR 72.124(a) are met. This ensures that when the LCO is applicable (upon introducing water into the MPC), the LCO will be met.

Surveillance Requirement 3.3.1.1 is modified by a note which states that SR 3.3.1.1 is only required to be performed if the MPC is submerged in water or if water is to be added to, or recirculated through the MPC. This reflects the underlying premise of this SR which is to ensure, once the correct boron concentration is established, it need only be verified thereafter if the MPC is in a state where the concentration could be changed. After the completion of the surveillance methods, events which might change the soluble boron concentration will be administratively controlled per the LCO. If actions are taken that could result in a reduction in the boron concentration the surveillance will be performed again.

There is no need to re-verify the boron concentration of the water in the MPC after it is removed from the spent fuel pool unless water is to be added to, or recirculated through the MPC, because these are the only credible activities that could potentially change the boron concentration during this time. This note also prevents the interference of unnecessary sampling activities while lid closure welding and other MPC storage preparation activities are taking place in an elevated radiation area atop the MPC. Plant procedures should ensure

BASES	
SURVEILLANCE REQUIREMENTS	that any water to be added to, or recirculated through the MPC is at a boron concentration greater than or equal to the minimum boron concentration specified in the LCO.
REFERENCES	1. HI-STORM FW FSAR Chapter 6.

CHAPTER 14†: QUALITY ASSURANCE PROGRAM

14.0 INTRODUCTION

14.0.1 Overview

This chapter provides a summary of the quality assurance program implemented by Holtec International for activities related to the design, qualification analyses, material procurement, fabrication, assembly, testing and use of structures, systems, and components of the Company's dry storage/transport systems including the HI-STORM UMAX System which includes the HI-TRAC transfer cask. This chapter is included in this FSAR to fulfill the requirements in 10 CFR 72.140 (c) (2) and 72.2(a)(1),(b).

Important-to-safety activities related to construction and deployment of the HI-STORM UMAX System are controlled under the NRC-approved Holtec Quality Assurance Program. The Holtec QA program manual [14.0.1] is approved by the NRC [14.0.2] under Docket 71-0784. The Holtec QA program satisfies the requirements of 10 CFR 72, Subpart G and 10 CFR 71, Subpart H. In accordance with 10 CFR 72.140(d), this approved 10 CFR 71 QA program will be applied to spent fuel storage cask activities under 10 CFR 72. The additional recordkeeping requirements of 10 CFR 72.174 are addressed in the Holtec QA program manual and must also be complied with.

The Holtec QA program is implemented through a hierarchy of procedures and documentation, listed below.

1. Holtec Quality Assurance Program Manual

2. Holtec Quality Assurance Procedures

3. a. Holtec Standard Procedures
   b. Holtec Project Procedures

Quality activities performed by others on behalf of Holtec are governed by the supplier's quality assurance program or Holtec's QA program extended to the supplier. The type and extent of Holtec QA control and oversight is specified in the procurement documents for the specific item or service being procured. The fundamental goal of the supplier oversight portion of Holtec's QA program is to provide the assurance that activities performed in support of the supply of safety-significant items and services are performed correctly and in compliance with the procurement documents.

† This chapter has been prepared in the format and section organization set forth in Regulatory Guide 3.61.

14.0.2 Graded Approach to Quality Assurance

Holtec International uses a graded approach to quality assurance on all safety-related or important-to-safety projects. This graded approach is controlled by Holtec Quality Assurance (QA) program documents as described in Subsection 14.0.1.

NUREG/CR-6407 [14.0.3] provides descriptions of quality categories A, B and C. Using the guidance in NUREG/CR-6407, Holtec International assigns a quality category to each individual, important-to-safety component of the HI-STORM UMAX System and HI-TRAC transfer cask. The ITS categories assigned to the HI-STORM UMAX cask components are identified in licensing drawing in Section 1.5. Quality categories for ancillary equipment are provided in Chapter 9 of this FSAR. Quality categories for other equipment needed to deploy the HI-STORM UMAX System at a licensee's ISFSI are defined on a case-specific basis considering the component's design function using the guidelines of NUREG/CR-6407 [14.0.3].

Activities affecting quality are defined by the purchaser's procurement contract for use of the HI-STORM UMAX System at an independent spent fuel storage installation (ISFSI) under the general license provisions of 10CFR72, Subpart K. These activities include any or all of the following: design, procurement, fabrication, handling, shipping, storing, cleaning, assembly, inspection, testing, operation, maintenance, repair and monitoring of HI-STORM UMAX structures, systems, and components (SSCs) that are important-to-safety.

The quality assurance program described in the QA Program Manual fully complies with the requirements of 10CFR72 Subpart G and the intent of NUREG-1536 [14.0.4]. However, NUREG-1536 does not explicitly address incorporation of a QA program manual by reference. Therefore, invoking the NRC-approved QA program in this FSAR constitutes a literal deviation from NUREG-1536. This deviation is acceptable since important-to-safety activities are implemented in accordance with the latest revision of the Holtec QA program manual and implementing procedures. Further, incorporating the QA Program Manual by reference in this FSAR avoids duplication of information between the implementing documents and the FSAR and any discrepancies that may arise from simultaneous maintenance to the two program descriptions governing the same activities.

14.1 REFERENCES

[14.0.1] Holtec International Quality Assurance Program, Latest Approved Revision on Docket 71-0784.

[14.0.2] NRC QA Program Approval for Radioactive Material Packages No. 0784, Docket 71-0784.

[14.0.3] NUREG/CR-6407, "Classification of Transportation Packaging and Dry Spent Fuel Storage System Components According to Importance to Safety," February 1996.

[14.0.4] NUREG-1536, "Standard Review Plan for Dry Cask Storage Systems," Rev 1, USNRC, 2010.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION IV
1600 EAST LAMAR BOULEVARD
ARLINGTON, TEXAS 76011-4511

December 19, 2018

EA-18-155

Mr. Doug Bauder
Vice President and Chief Nuclear Officer
Southern California Edison Company
San Onofre Nuclear Generating Station
P.O. Box 128
San Clemente, CA 92674-0128

SUBJECT: ERRATA: SAN ONOFRE NUCLEAR GENERATING STATION - NRC SPECIAL INSPECTION REPORT 050-00206/2018-005, 050-00361/2018-005, 050-00362/2018-005, 072-00041/2018-001 AND NOTICE OF VIOLATION

Mr. Bauder:

It was identified that the U.S. Nuclear Regulatory Commission (NRC) Special Inspection Report No. 050-00206/2018-005, 050-00361/2018-005, 050-00362/2018-005, 072-00041/2018-001, dated November 28, 2018 Agency Document and Management System (ADAMS) (ADAMS Accession No. ML18332A357) and Notice of Violation (Notice) incorrectly identified the cited violation against 10 CFR 72.192, regarding “Operator training and certification program,” in lieu of citing the violation against 10 CFR 72.190, “Operator requirements.” The corrected inspection report and Notice shall refer to “10 CFR 72.190” in all applicable areas. As specified in 10 CFR 72.13, the regulation identified under 10 CFR 72.190 is applicable to a general licensee, which is the type of license held by Southern California Edison Company. The inspection report and all its enclosures, including the Notice, is reissued in its entirety under the same inspection report number and is enclosed.

The change to the citation in the Notice involving training and certification of personnel does not change the content of the inspection report, or the two apparent violations. As such, the communications provided in the November 28, 2018, inspection report regarding your opportunity to request a predecisional enforcement conference (PEC) or alternative dispute resolution (ADR) remains in effect from the date of the original inspection report, November 28, 2018. On December 10, 2018, SONGS informed the NRC that it requested a PEC. My staff is working with your staff to schedule the PEC.

In accordance with 10 CFR 2.390 of the NRC's "Agency Rules of Practice and Procedure," a copy of this letter, its enclosures, and your response (if any), will be made available electronically for public inspection in the NRC Public Document Room and from the NRC's ADAMS, accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html.

If you have any questions concerning this matter, please contact Dr. Janine F. Katanic, CHP, of my staff at 817-200-1151.

Sincerely,

/RA/

Troy W. Pruett, Director
Division of Nuclear Materials Safety

Docket Nos.: 50-206; 50-361; 50-362; 72-041
License Nos.: NPF-10; NPF-15; DPR-13

Enclosure:
Revised NRC Special Inspection
Report 050-00206/2018-005,
050-00361/2018-005, 050-00362/2018-005,
and 072-00041/2018-001

ERRATA: SAN ONOFRE NUCLEAR GENERATING STATION - NRC SPECIAL INSPECTION REPORT 050-00206/2018-005, 050-00361/2018-005, 050-00362/2018-005, 072-00041/2018-001 AND NOTICE OF VIOLATION – DATED DECEMBER 19, 2018

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AND REVISED NOTICE OF VIOLATION
(ML18341A172)

Enclosure

UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION IV
1600 EAST LAMAR BOULEVARD
ARLINGTON, TEXAS 76011-4511

December 19, 2018

EA-18-155

Mr. Doug Bauder
Vice President and Chief Nuclear Officer
Southern California Edison Company
San Onofre Nuclear Generating Station
P.O. Box 128
San Clemente, CA 92674-0128

SUBJECT: REVISED NRC SPECIAL INSPECTION REPORT 050-00206/2018-005, 050-00361/2018-005, 050-00362/2018-005, 072-00041/2018-001 AND REVISED NOTICE OF VIOLATION

Mr. Bauder:

This letter refers to the Special Inspection conducted on September 10-14, 2018, at your facility in San Clemente, California. The inspection was conducted in response to the misalignment of a loaded spent fuel storage canister as it was being downloaded into the storage vault at the San Onofre Nuclear Generating Station (SONGS). Based on the criteria specified in Management Directive 8.3, "NRC Incident Investigation Program," the Nuclear Regulatory Commission (NRC) initiated a Special Inspection in accordance with Inspection Procedure 93812, "Special Inspection." The basis for initiating the Special Inspection and the focus areas for review are detailed in the Special Inspection Charter (Enclosure 3), dated August 17, 2018 (Agencywide Document Access and Management System (ADAMS) Accession ML18229A203).

The enclosed report documents the results of the inspection. The inspectors discussed the preliminary inspection findings with Mr. Thomas Palmisano and members of your staff on September 14, 2018, at the conclusion of the onsite portion of the inspection. A final exit briefing was conducted telephonically with Mr. Palmisano and members of your staff on November 1, 2018.

Based on the results of the Special Inspection, two apparent violations were identified and are being considered for escalated enforcement action in accordance with the NRC Enforcement Policy. The current Enforcement Policy is included on the NRC Web site at http://www.nrc.gov/about-nrc/regulatory/enforcement/enforce-pol.html. The circumstances surrounding these apparent violations, the significance of the associated issues, and the need for corrective actions were discussed with Mr. Palmisano at the conclusion of the onsite inspection and during the final telephonic exit briefing. The apparent violations involved the failure to: (1) ensure important-to-safety equipment was available to provide redundant drop protection features for a spent fuel canister during downloading operations; and (2) make a timely notification to the NRC Headquarters Operations Center for the August 3, 2018, disabling of important-to-safety equipment.

The NRC is concerned about apparent weaknesses in management oversight of the dry cask storage operations. Your staff did not perform adequate direct observational oversight of downloading activities performed by your contractor, ensure adequate training of individuals responsible for performing downloading operations, provide adequate procedures for downloading operations, or ensure that conditions adverse to quality were entered into the corrective action program. The NRC identified that a causal factor for the misalignment incident involved management weakness in the oversight of dry cask storage operations.

Before the NRC makes its enforcement decision, we are providing you with an opportunity to: (1) request a predecisional enforcement conference (PEC) or (2) request alternative dispute resolution (ADR). If a PEC is held, it will be open for public observation and the NRC will issue a press release to announce the time and date of the conference.

If you choose to request a PEC, the conference will afford you the opportunity to provide your perspective on these matters and any other information that you believe the NRC should take into consideration before making an enforcement decision. The decision to hold a PEC does not mean that the NRC has determined that a violation has occurred or that enforcement action will be taken. This conference would be conducted to obtain information to assist the NRC in making an enforcement decision.

The topics discussed during the conference may include information to determine whether a violation occurred, information to determine the significance of a violation, information related to the identification of a violation, and information related to any corrective actions taken or planned. In presenting your corrective actions, you should be aware that the promptness and comprehensiveness of your actions will be considered in assessing any civil penalty for the apparent violations. The guidance in NRC Information Notice 96-28, “Suggested Guidance Relating to Development and Implementation of Corrective Action,” may be helpful and can be obtained at the NRC Web site at http://pbadupws.nrc.gov/docs/ML0612/ML061240509.pdf.

In lieu of a PEC, you may also request ADR with the NRC in an attempt to resolve this issue. Alternative dispute resolution is a general term encompassing various techniques for resolving conflicts using a neutral third party. The technique that the NRC has decided to employ is mediation. Mediation is a voluntary, informal process in which a trained neutral mediator works with parties to help them reach resolution. If the parties agree to use ADR, they select a mutually agreeable neutral mediator who has no stake in the outcome and no power to make decisions. Mediation gives parties an opportunity to discuss issues, clear up misunderstandings, be creative, find areas of agreement, and reach a final resolution of the issues.

Additional information concerning the NRC’s program can be obtained at http://www.nrc.gov/about-nrc/regulatory/enforcement/adr.html. The Institute on Conflict Resolution at Cornell University has agreed to facilitate the NRC’s program as a neutral third party. Please contact the Institute on Conflict Resolution at 877-733-9415 within 10 days of the date of this letter if you are interested in pursuing resolution of these issues through ADR. Alternative dispute resolution sessions are not conducted with public observation though the outcome of the ADR agreement is made public.

A PEC should be held within 30 days and an ADR session within 45 days of the date of this letter. Please contact Dr. Janine F. Katanic at 817-200-1151 within 10 days of the date of this letter to notify the NRC of your intended response.

In addition, please be advised that the number and characterization of apparent violations described in the enclosed inspection report may change as a result of further NRC review. You will be advised by separate correspondence of the results of our deliberations on this matter.

The NRC determined that three Severity Level IV violations of NRC requirements occurred. These violations were evaluated in accordance with Section 2.2.2 of the NRC Enforcement Policy. The NRC determined the issuance of a Notice of Violation (Notice) is appropriate because the actions to restore compliance have not been fully developed and implemented, and the actions must be effective prior to beginning fuel handling activities.

The three Severity Level IV violations are cited in the enclosed Notice and the circumstances surrounding them are described in detail in the subject inspection report. The violations involved failures to: (1) identify conditions potentially adverse to quality for placement into your corrective actions program; (2) assure that operations of important to safety equipment were limited to trained and certified personnel or under direct supervision; and (3) provide adequate procedures for dry cask storage operations involving downloading operations.

In accordance with 10 CFR 2.390 of the NRC's "Agency Rules of Practice and Procedure," a copy of this letter, its enclosures, and your response, will be made available electronically for public inspection in the NRC Public Document Room and from the NRC's ADAMS, accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html. To the extent possible, your response should not include any personal privacy or proprietary information so that it can be made available to the public without redaction.

If you have any questions concerning this matter, please contact Dr. Janine F. Katanic, CHP, of my staff at 817-200-1151.

Sincerely,

/RA/

Troy W. Pruett, Director
Division of Nuclear Materials Safety

Docket Nos.: 50-206; 50-361; 50-362; 72-041
License Nos.: NPF-10; NPF-15; DPR-13

Enclosures:
1. Notice of Violation
2. Revised NRC Special Inspection Report 050-00206/2018-005, 050-00361/2018-005, 050-00362/2018-005, and 072-00041/2018-001
3. Special Inspection Charter dated August 17, 2018 (ML18229A203)

NOTICE OF VIOLATION

Southern California Edison Company
San Clemente, CA

Docket Nos.: 050-00206, 050-00361, 050-00362, 072-00041
License Nos.: NPF-10; NPF-15; DPR-13
EA No: 18-155

During an NRC Special Inspection conducted September 10 through November 1, 2018, three violations of NRC requirements were identified. In accordance with the NRC Enforcement Policy, the violations are listed below:

A. 10 CFR 72.172 requires, in part, that, licensees establish measures to ensure that conditions adverse to quality, such as failures, malfunctions, deficiencies, and deviations, are promptly identified and corrected.

   Contrary to the above, from January 30 to August 3, 2018, the licensee failed to establish measures to ensure that conditions adverse to quality, such as failures, malfunctions, deficiencies, and deviations, were promptly identified and corrected. Specifically:

   1. On July 22, 2018, the loading crew experienced difficulty in aligning canister 28 for downloading into the independent spent fuel installation vault. However, the licensee failed to enter this deviation in downloading conditions into its corrective action program to determine the cause of the misalignment problem and develop corrective actions to preclude reoccurrence.

   2. From January 30 to August 3, 2018, during canister downloading, contact between the canister and vault components frequently occurred. However, the licensee failed to enter instances of contact into its corrective action program and perform an assessment to disposition the exterior conditions of the downloaded canisters and vault components.

This is a Severity Level IV violation (NRC Enforcement Policy Section 6.3).

B. 10 CFR 72.190 requires, in part, that the operation of equipment and controls that have been identified as important to safety in the Safety Analysis Report and in the license must be limited to trained and certified personnel or be under the direct supervision of an individual with training and certification in the operation. The HI-STORM UMAX SYSTEM Final Safety Analysis Report (FSAR), Revision 4, dated August 14, 2017, specifies, in part, that the operations at the independent spent fuel storage installation are governed by the HI-STORM FW SYSTEM FSAR, Revision 5, dated June 20, 2017, which specifies that the multipurpose canister lifting slings and multipurpose canister lift attachments are designated as important to safety equipment.

   Contrary to the above, from January 30 to August 3, 2018, the licensee failed to assure that operations of equipment and controls that had been identified as important to safety in the Safety Analysis Report were limited to trained and certified personnel or were under the direct supervision of an individual with training and certification in the operation. Specifically:

1. The training program failed to adequately train and certify the rigger/spotter position involved in the important to safety downloading operation.

2. The training program for the vertical cask transporter operator position failed to have adequate proficiency testing, on the controls related to the load indicating device and downloading operations.

This is a Severity Level IV violation (NRC Enforcement Policy Section 6.3).

C. 10 CFR 72.150, requires, in part, that the licensee prescribe activities affecting quality by documented instructions or procedures of a type appropriate to the circumstances and must include appropriate quantitative or qualitative acceptance criteria for determining that important activities have been satisfactorily accomplished.

Contrary to the above, from January 30 to August 3, 2018, the licensee failed to prescribe activities affecting quality by documented instructions or procedures of a type appropriate to the circumstances and include appropriate quantitative or qualitative acceptance criteria for determining that important activities have been satisfactorily accomplished. Specifically:

1. Procedure HPP-2464-400, "Multi-Purpose Canister Transfer at SONGS," Revision 15, step 7.6.23, failed to provide qualitative and quantitative directions for the vertical cask transporter operator to monitor control panel indications that would identify a canister had become misaligned during downloading operation.

2. Procedure HPP-2464-400, "Multi-Purpose Canister Transfer at SONGS," Revision 15, step 7.6.23, failed to include adequate instructions for the rigger/spotter to monitor the downloading slings for a slack condition.

This is a Severity Level IV violation (NRC Enforcement Policy Section 6.3).

Pursuant to the provisions of 10 CFR 2.201, Southern California Edison Company is hereby required to submit a written statement or explanation to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-0001, with a copy to the Regional Administrator, U.S. Nuclear Regulatory Commission, Region IV, 1600 E. Lamar Blvd., Arlington, TX 76011, within 30 days of the date of the letter transmitting this Notice of Violation (Notice).

This reply should be clearly marked as a "Reply to a Notice of Violation, EA-18-155" and should include, for each violation: (1) the reason for the violation, or, if contested, the basis for disputing the violation or severity level; (2) the corrective steps that have been taken and the results achieved; (3) the corrective steps that will be taken; and (4) the date when full compliance will be achieved. Your response may reference or include previous docketed correspondence, if the correspondence adequately addresses the required response. If an adequate reply is not received within the time specified in this Notice, an order or a Demand for Information may be issued requiring information as to why the license should not be modified, suspended, or revoked, or why such other action as may be proper should not be taken. Where good cause is shown, consideration will be given to extending the response time.

If you contest this enforcement action, you should also provide a copy of your response, with the basis for your denial, to the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001.

Your response will be made available electronically for public inspection in the NRC Public Document Room or in the NRC's Agencywide Documents Access and Management System (ADAMS), accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html. To the extent possible, your response should not include any personal privacy or proprietary information so that it can be made available to the public without redaction.

If personal privacy or proprietary information is necessary to provide an acceptable response, then please provide a bracketed copy of your response that identifies the information that should be protected and a redacted copy of your response that deletes such information. If you request withholding of such material, you must specifically identify the portions of your response that you seek to have withheld and provide in detail the bases for your claim of withholding (e.g., explain why the disclosure of information will create an unwarranted invasion of personal privacy or provide the information required by 10 CFR 2.390(b) to support a request for withholding confidential commercial or financial information).

Dated this 19th day of December 2018

U.S. NUCLEAR REGULATORY COMMISSION
REGION IV

Docket Nos.:	50-206; 50-361; 50-362; 72-041
License Nos.:	NPF-10; NPF-15; DPR-13
Report No.:	050-00206/2018005; 050-00361/2018005; 050-00362/2018005; and 072-00041/2018001
Enterprise Identifier:	I-2018-001-0138
EA No.:	18-155
Licensee:	Southern California Edison Company
Location:	San Clemente, CA 92674-012
Inspection Dates:	Onsite September 10-14, 2018 In-office review through November 1, 2018
Exit Meeting Date:	November 1, 2018
Inspectors:	Eric Simpson, CHP, Health Physicist Fuel Cycle and Decommissioning Branch Division of Nuclear Materials Safety, Region IV Marlone Davis, Senior Inspector Inspections and Operations Branch Division of Spent Fuel Management W. Chris Smith, Reactor Inspector Engineering Branch 1 Division of Reactor Safety, Region IV
Accompanied By:	Janine F. Katanic, PhD, CHP, Chief Fuel Cycle and Decommissioning Branch Division of Nuclear Materials Safety, Region IV Patricia Silva, Chief Inspections and Operations Branch Division of Spent Fuel Management Troy W. Pruett, Director Division of Nuclear Materials Safety, Region IV
Approved By:	Troy W. Pruett, Director Division of Nuclear Materials Safety, Region IV
Attachment:	Supplemental Inspection Information

Enclosure 2

EXECUTIVE SUMMARY

NRC Special Inspection Report 050-00206/2018005; 050-00361/2018005; 050-00362/2018005; and 072-00041/2018-001

On September 10-14, 2018, the U.S. Nuclear Regulatory Commission performed an announced Special Inspection of the independent spent fuel storage installation at the decommissioning San Onofre Nuclear Generating Station in San Clemente, California. The inspection continued with an in-office review of training material, licensee analyses, procedures, and other materials gathered during the onsite inspection through November 1, 2018. The Southern California Edison Company, the licensee and owner of San Onofre Nuclear Generating Station, has an NRC General License for its independent spent fuel installation under Title 10 of the *Code of Federal Regulations* (10 CFR) Part 72. The scope of the inspection was to evaluate the facts and circumstances involved in the August 3, 2018, misalignment incident, and review the licensee's follow-up investigation, causal evaluation, and planned corrective actions.

<u>NRC Special Inspection of San Onofre Nuclear Generating Station Canister Misalignment Incident of August 3, 2018</u>

- The licensee's actions that led to disabling the important to safety downloading slings and removal of redundant drop protection features were identified as an apparent violation of Technical Specification 5.2.c.3 requirements. (Section 3.1.1)

- The NRC team identified missed opportunities where the licensee could have addressed the potential for a downloading misalignment. For example, on July 22, 2018, one of the crews experienced misalignment difficulty resulting in a prolonged downloading operation. The licensee did not enter the adverse condition into the corrective action program to determine the cause and develop appropriate corrective actions. This was identified as a Severity Level IV violation of 10 CFR 72.172 requirements. (Section 3.1.1)

- Personnel lacked the proper training, proficiency testing, and certifications to operate important to safety equipment identified in the HI-STORM UMAX SYSTEM Final Safety Analysis Report, Revision 4, dated August 14, 2017. This was identified as a Severity Level IV violation of 10 CFR 72.190 requirements. (Section 3.1.2)

- Dry cask storage procedures did not provide adequate directions for how to determine the downloader slings were slack. Slack in the slings was an indicator of a loss-of-load. Further, procedures did not include qualitative or quantitative means to determine when a canister had become misaligned. These procedure inadequacies were identified as a Severity Level IV violation of 10 CFR 72.150 requirements. (Section 3.1.3)

- No licensee or contractor oversight staff were in direct visual observation of important to safety activities during downloading operations on August 3, 2018. Licensee oversight was not a part of communications between the cask loading supervisor, the rigger/spotter, and vertical cask transporter operator during downloading operations. (Section 3.1.3)

- The licensee concluded and the NRC agreed that the minor removal of divider shell coating during downloading operations did not affect the design functions for shielding, structural, and thermal safety functions. The licensee's plan to address future inspection of the divider shells in their aging management program is acceptable. (Section 3.1.4)

- The licensee failed to make the required 24-hour NRC notification of the August 3, 2018, incident where important to safety equipment was disabled when required to mitigate the consequences of an accident and no redundant equipment was available to perform the safety function. This failure was identified as an apparent violation of 10 CFR 72.75(d) requirements. (Section 3.1.4)

- The causal evaluations performed by the licensee and its contractor identified apparent and root causes for the August 3, 2018, canister misalignment incident that included inadequate training, inadequate procedures, poor utilization of the corrective action program, and insufficient management oversight. (Section 3.1.5)

- The licensee's consequence analysis resulting from a hypothetical 25-foot canister drop determined that the canister integrity would be maintained. The NRC will continue to inspect the licensee's consequence analysis. (Section 3.1.5)

- The licensee provided an analysis to demonstrate that wear on canister 29 during the downloading incident would meet established acceptance criteria. The NRC determined that more analysis was required to accept that the canister meets design requirements. This charter item will be reviewed during a future NRC inspection. (Section 3.1.6)

- All associated corrective actions for the August 3, 2018, incident had not been fully developed and implemented by the licensee. The NRC will review the licensee's revised procedures, training plans, equipment modifications, and performance testing (dry runs) of its dry cask storage operations during a future inspection to determine the effectiveness of corrective actions for the incident. (Section 3.1.7)

REPORT DETAILS

1 Inspection Scope

On September 10-14, 2018, the NRC performed an announced Special Inspection at the San Onofre Nuclear Generating Station (SONGS) in San Clemente, California, which was followed by in-office reviews of additional information provided by the licensee through November 1, 2018. The scope of the inspection was to interview personnel associated with the August 3, 2018, misalignment incident to independently evaluate the circumstances of the canister misalignment; identify and review all pertinent records, documents, and procedures related to the licensee's downloading operations; evaluate procedure adequacy and adherence; evaluate the reportability requirements; and to evaluate the root cause analyses and corrective actions to prevent recurrence.

2 Background

2.1 General Description of Multi-purpose Canister Downloading Operations

On November 8, 2018, the NRC conducted a public meeting webinar (NRC's Agencywide Documents Access and Management System (ADAMS) Accession ML18319A139). The presentation provides a summary of a downloading operation.

A vertical cask transporter (VCT) is used for transporting the transfer cask and multi-purpose canister (MPC or canister) loaded with spent fuel onto the independent spent fuel storage installation (ISFSI) pad. Dry cask storage workers manipulate the VCT to align the transfer cask over the ISFSI vertical ventilated module (VVM or vault) in which the canister will be stored. Once alignment has been achieved and the transfer cask is securely bolted to a mating device, the transfer cask is disconnected from the VCT. Lifting slings are connected to the top of the canister and the VCT overhead lift beam. The VCT lift beam is raised until the load of the canister is supported and no longer resting on the bottom of the transfer cask.

While the canister is being supported by the lift beam and slings, a drawer on the mating device is opened. Once the drawer is open, the VCT operator lowers the lift beam, which lowers the canister into the storage vault. The VCT can be moved during the download to make fine adjustments for canister alignment within the vault. While the canister is being lowered, it passes through a divider shell assembly. The divider shell has a shield ring that the canister must pass through as it is being lowered into the vault. When fully downloaded, the canister will be seated on a pedestal in the cavity enclosure container in the vault.

2.2 August 3, 2018 Canister Misalignment

On August 3, 2018, as the loaded canister was being lowered into the vault, personnel failed to notice that the canister was misaligned. The licensee and its contractor continued to lower the VCT lift beam until staff believed that the canister had been fully lowered to the bottom of the vault. Staff involved in the download failed to recognize the lifting slings were slack. A radiation protection technician identified radiation readings that were not consistent with a fully lowered canister. The licensee then identified that the loaded spent fuel canister was resting on a shield ring near the top of the vault,

preventing it from being lowered, and that the rigging and lifting slings were slack and no longer bearing the load of the canister.

With the slings slack, the lifting equipment was no longer capable of performing it's important to safety function of holding and controlling the loaded canister. The canister could have experienced an approximately 17-18 foot drop into the storage vault if the canister had slipped off the shield ring. This load drop accident is not a condition analyzed in the dry fuel storage system's Final Safety Analysis Report (FSAR).

The licensee restored the control of the load to the slings and lifting devices. The estimated time the canister was in an unsupported position was approximately 45 minutes. The licensee repositioned and lowered the canister into the vault. The licensee subsequently halted all dry fuel storage movement operations in order to fully investigate the incident and develop corrective actions to prevent recurrence.

The licensee informed Region IV staff of the misalignment incident on August 6, 2018. Region IV discussed the licensee's plans for evaluation and follow-up for the incident and the status of fuel loading operations. The licensee agreed to suspend fuel loading operations until such time as their senior management was satisfied with their corrective actions, the NRC completed their inspection, and the NRC determines that corrective actions are sufficient to prevent a similar occurrence. Region IV chartered a Special Inspection Team to review the incident, any relevant background information, causal and risk assessments conducted by the licensee, and proposed and completed corrective actions.

3 Special Inspection Charter (IP 93812)

3.1 Inspection Scope

Following the notification to NRC Region IV of the August 3, 2018, misalignment incident, the NRC evaluated the information provided against the criteria for a reactive inspection. Based on the criteria in Management Directive 8.3, "NRC Incident Investigation Program," and Inspection Manual Chapter 0309, "Reactive Inspection Decision Basis for Reactors," a decision was made to perform a Special Inspection. The Special Inspection Charter is provided in Enclosure 3.

The Special Inspection was conducted onsite from September 10-14, 2018, and continued with in-office review until November 1, 2018. The Special Inspection focused on understanding the August 3, 2018, misalignment incident. The inspection included interviewing personnel involved in the incident, developing a timeline, and assessing the licensee's immediate corrective actions.

The sections below provide inspection details for each of the Special Inspection Charter items.

3.1.1 Charter Item 5

Inspection Scope

"Interview personnel associated with the August 3, 2018, misalignment incident to develop a timeline to ensure the licensee's investigation contained all necessary information to identify all contributing factors and develop adequate corrective actions."

The NRC team interviewed licensee and contractor staff involved or present during the August 3, 2018, misalignment incident. The NRC also reviewed records related to dry cask storage operations.

Observations and Findings

Based on interviews and records reviewed, the following timeline was developed:

Date/Time (± 30 minutes)	Activity
August 3, 2018	
12:40 p.m.	Downloading begins for canister 29: All dry cask storage supervision and licensee oversight, including radiation protection staff exited the ISFSI pad to stand in a low-dose area on the ISFSI pad ramp (approximately 150 feet away from the operations). Only the rigger/spotter in the motor-powered boom lift device man-basket (JLG) and the VCT operator remained on the ISFSI pad.
1:05 p.m.	VCT operator and rigger/spotter notify cask loading supervisor (CLS) that the canister has been fully lowered into the ISFSI vault.
1:12 p.m.	The radiation protection technician (RPT) determines radiation levels indicate that the canister was not fully lowered. Work activities were stopped to plan recovery actions with the radiation protection supervisor and CLS. The rigger in charge (RIC) began making preparations to enter the JLG.

1:15 p.m.	Notifications were made to Holtec management.
	The RIC was escorted to the JLG by an RPT.
	The RIC recognized the downloading slings were slack and bundled on the ground near the base of the VCT.
1:33 p.m.	The RIC observed the top of the canister was about 4 feet from the top of the transfer cask and not lowered into the vault.
	The RIC directed the VCT operator to lift the canister.
1:41 p.m.	The canister load was fully supported by the VCT and downloading slings.
1:50 p.m.	An alternate CLS arrived and began to direct operations for downloading to the VCT operator.
	The alternate CLS and RIC noted that during downloading operations the canister experienced interference twice and had to be re-aligned.
2:22 p.m.	Downloading operations completed.
6:00 p.m.	Licensee places hold on all lifting operations.
August 6, 2018	At approximately 4 pm (CDT), the licensee informally contacted NRC Region IV to discuss the August 3, 2018, misalignment incident.
August 7, 2018	NRC Region IV and licensee management agreed that ISFSI operations would cease until the NRC performed an inspection and reviewed the licensee’s corrective actions to resume work.
September 14, 2018	At 4 pm (ET) the licensee made a formal notification per 10 CFR 72.75(d)(1) to the NRC Headquarters Operations Center regarding the August 3, 2018, misalignment incident.

Violation of 10 CFR 72.172, Corrective Actions

Interviews with Williams Industrial Services Group and Sonic Systems (Holtec International subcontractors) employees indicated that of a loss-of-load condition or a canister misalignment issue was experienced during dry run evolutions and known to several dry cask storage workers. The Special Inspection team identified a prior canister misalignment issue that occurred on July 22, 2018, in which downloading operations lasted 90 minutes, instead of the expected 15 minutes for downloading canister 28. This incident was documented in a Production Traveler. A Production Traveler is a document that the licensee uses to track the performance of dry fuel storage operations by the

contractor, Holtec International. The Production Travelers were used to track how well the contractor was providing their contracted services to the licensee. The licensee did not enter this condition adverse to quality into its corrective action program.

Licensee oversight generally waited for Holtec staff to initiate a field condition report (FCR) before writing a corresponding condition report. In the Production Traveler for canister 28, the 90 minute delay was related to adjustments that were needed for the VCT towers as canister weight started to lower prematurely before the downloading was complete. This type of misalignment also occurred during the August 3, 2018, incident. On July 22, 2018, the downloading crew for canister 28, noted the reduction in the canister weight and corrected the alignment error. The canister was never unsupported by the slings. No condition report or FCR was generated by either the licensee or contractor.

Through interviews with licensee and contractor staff, the NRC determined that between January 30 and August 3, 2018, the downloading activity often involved contact between the canister and other vault components during downloading. The licensee and its contractor did not enter the misalignment and contact events into the corrective action program. Consequently, actions to assess and disposition the exterior conditions of the downloaded canisters and other components within the vault, such as the divider shell assembly, were not performed. The licensee is responsible to ensure the important to safety components continue to meet their original design criteria and address any aging management concerns the changes could impact. Any deviations, such as scratches or removal of coatings are required to be evaluated to ensure the deviations are not detrimental to the system.

Interviews with individuals involved in dry cask loading operations in August 2018, revealed that the difficulty in aligning the canister was not shared with others, nor was it incorporated into procedures or formal training programs. The VCT operator and the rigger/spotter in charge of downloading operations during the August 3, 2018, incident indicated that they did not know until afterwards that the condition they experienced was something that should have been anticipated.

Title 10 CFR 72.172 requires, in part, that, licensees establish measures to ensure that conditions adverse to quality, such as failures, malfunctions, deficiencies, and deviations are promptly identified and corrected. Contrary to the above, the licensee failed to establish measures to ensure that conditions adverse to quality, such as failures, malfunctions, deficiencies, and deviations were promptly identified and corrected. Specifically:

1. On July 22, 2018, the crew experienced difficulty in aligning canister 28 for downloading into the ISFSI vault. However, the licensee failed to enter this deviation in downloading conditions into its corrective action program to determine the cause of the misalignment problem and develop corrective actions to preclude reoccurrence.

2. From January 30 to August 3, 2018, during canister downloading, contact between the canister and the vault components frequently occurred. The licensee failed to enter instances of contact into its corrective action program and perform an assessment to disposition the exterior conditions of the downloaded canisters and vault components.

The team determined that this violation was more than minor because the failure to implement corrective actions contributed to the misalignment incident of August 3, 2018. Additionally, the failure to evaluate and disposition wear marks on a canister, if left uncorrected, could impact the adequacy of the aging management program. The Special Inspection team assessed and dispositioned this violation in accordance with Section 2.2.2 of the NRC Enforcement Policy. The team characterized the violation as a Severity Level IV violation. The NRC determined the issuance of a Notice is appropriate because the actions to restore compliance have not been fully developed and implemented, and the actions must be effective prior to beginning fuel handling activities. (VIO 07200041/2018-001-01, Failure to identify and correct conditions adverse to quality)

Apparent Violation of Technical Specification 5.2.c.3, Redundant Lifting Equipment

On August 3, 2018, the licensee performed operations involving movement of a loaded spent fuel storage canister into its ISFSI vault. As the loaded spent fuel canister was being lowered into the vault, licensee and contractor personnel failed to notice that the canister was misaligned and the weight of the canister was not being supported by the redundant important to safety slings (See Sections 2.1 and 2.2).

Title 10 CFR 72.212(b)(3) requires, in part, that each cask used by the general licensee conforms to the terms, conditions, and specifications of a Certificate of Compliance listed in 10 CFR 72.214. Title 10 CFR 72.214 includes a list of all the approved spent fuel storage casks that can be utilized under the conditions specified in a specific Certificate of Compliance, including Amendment 2 of Certificate of Compliance 072-01040. Certificate of Compliance 072-01040, Amendment 2, Condition 4, “HEAVY LOADS REQUIREMENTS,” requires that lifting operations outside of structures governed by 10 CFR Part 50 must be in accordance with Technical Specifications, Appendix A, Section 5.2.

Technical Specification, Appendix A, Section 5.2.c.3 requires that the transfer cask, when loaded with spent fuel, may be lifted and carried at any height during multi-purpose canister transfer operations provided the lifting equipment is designed with redundant drop protection features which prevent uncontrolled lowering of the load.

Contrary to the above, on August 3, 2018, the licensee failed to ensure that redundant drop protection features were available to prevent uncontrolled lowering of the load. Specifically, the licensee inadvertently disabled the redundant important to safety downloading slings while lowering canister 29 into the storage vault. During the approximately 45 minute time-frame, the canister rested on a shield ring unsupported by the redundant downloading slings at approximately 17-18 feet above the fully seated position. This failure to maintain redundant drop protection placed canister 29 in an unanalyzed condition because the postulated drop of a loaded spent fuel canister is not analyzed in the FSAR.

The licensee’s failure to ensure the system’s designed redundant drop protection features were available to prevent uncontrolled lowering of the loaded canister was identified as an apparent violation of Technical Specification 5.2.c.3.
(AV 07200041/2018-001-02, Failure to ensure redundant drop protection features are available)

Conclusions

The licensee failed to adequately implement the corrective action program for ISFSI operations. This failure resulted in missed opportunities to resolve misalignment errors during canister downloading operations between January 30 and August 3, 2018, and a violation of 10 CFR 72.172.

On August 3, 2018, the licensee failed to recognize that a misalignment of a canister during downloading operations caused redundant drop protection (slings) to be disabled and an apparent violation of Technical Specification 5.2.c.3.

3.1.2 Charter Item 1

Inspection Scope

"Identify and review all pertinent records, documents, and procedures related to the licensee's downloading operations at the ISFSI pad including but not limited to: worker training and qualifications; rigging equipment qualification, testing, and preventative maintenance; and lifting equipment qualification, testing, and preventative maintenance. Evaluate the adequacy of the above noted procedures, worker training, and equipment testing and preparation."

The Special Inspection team reviewed licensee rigging procedures and NUREG-0612 "Control of Heavy Loads at Nuclear Power Plants," training modules. The team reviewed the qualifications for the dry cask storage workers including the records for the workers involved in the August 3, 2018, misalignment incident. The team reviewed the inspection and maintenance records for special lifting devices used during dry fuel storage operations and the qualification records for rigging equipment. The team reviewed procedures used during canister downloading operations.

Observations and Findings

The equipment used for dry cask storage operations met applicable inspection requirements specified in the Holtec HI-STORM UMAX FSAR. The special lifting devices used to transport the transfer cask and to perform downloading operations were designed and tested according to American National Standards Institute (ANSI) N14.6, "American National Standard for Radioactive Materials – Special Lifting Devices for Shipping Containers Weighing 10,000 Pounds or More." The slings used during downloading had a sufficient load rating for the maximum credible load imposed by the canister. The slings were tested according to the safety requirements of American Society of Mechanical Engineers (ASME) B30.9, "Slings." The purchase specifications, qualifications, and maintenance records for the VCT, downloading slings, canister lift cleats, lift lugs, and lift links were satisfactory.

Violation of 10 CFR 72.190, Training and Certification Qualifications

The NRC team reviewed the qualifications of workers involved in the August 3, 2018, incident. Interviews with the individuals primarily responsible for verifying that the canister was properly downloaded into the ISFSI vault showed that the licensee's training program was inadequate for the positions that are designated as rigger/spotter and VCT operator. The VCT operator training program qualifications did not establish

adequate required proficiency training exercises for downloading operations. The VCT operator on August 3, 2018, had never been tested on or exercised with the canister simulator during a pre-operational testing, “dry run” downloading operation. The August 3, 2018, misalignment incident was the first time the VCT operator had actually completed downloading operations as the VCT operator.

Neither the rigger/spotter nor VCT operator was properly trained in determining a loss-of-load condition during downloading operations. The VCT operator stated that he was knowledgeable of the VCT human-machine interface (HMI) screens and that indications provided a digital reading that could allow the operator to determine if the canister was not supported by the slings. However, the VCT operator stated that he did not use the VCT HMI screen to monitor the load of the canister at any time during the August 3, 2018, downloading operations. The VCT operator indicated that he only utilized the HMI screen to determine how evenly the VCT lift beam was descending.

From his position on the VCT, the VCT operator could not see the canister downloader slings. The only indication of a loss-of-load would come from monitoring the VCT hydraulic beam pressure digital reading on the VCT HMI screen, which was not performed. Since the operator had not performed any proficiency training with the VCT during a dry run downloading operation, the individual was inexperienced with the use of the HMI screen to monitor load loss.

The licensee’s training program did not provide a formal process to be qualified for the rigger/spotter position during downloading operations. The rigger/spotter stated that he was not trained on and did not know his roles and responsibilities during the downloading evolution. The August 3, 2018, misalignment incident was the first time the rigger/spotter had attempted to perform downloading operations as the rigger/spotter in the JLG.

The NRC team’s interview with the foreman indicated that the rigger/spotter was selected primarily because of his low accumulated radiation dose. From interviews with licensee and contractor staff, an experienced RIC was usually the individual placed in the JLG and acted as the rigger/spotter for the downloading operations. On August 3, 2018, it was the RIC who eventually entered the JLG after the misalignment and directed the VCT operator to lift the canister with the VCT lift beam to regain the load on the slings. The RIC had immediately recognized that the canister was not downloaded into the ISFSI vault when he arrived and saw the condition of the downloader slings.

The failure to ensure operators are adequately qualified and proficiency tested when operating important to safety equipment and directing critical lift operations is a performance deficiency. The licensee’s training program that allowed the rigger/spotter and VCT operator to be placed into a situation where their lack of training rendered them incapable of meeting the requirements for the job represented a failure of the licensee’s training program.

Title 10 CFR 72.190 requires, in part, that the operation of equipment and controls that are identified as important to safety in the Safety Analysis Report must be limited to trained and certified personnel or be under the direct supervision of an individual with training and certification in the operation. The HI-STORM UMAX SYSTEM FSAR, Revision 4, dated August 14, 2017, specifies, in part, that the operations at the ISFSI are

governed by the HI-STORM FW SYSTEM FSAR, Revision 5, dated June 20, 2017, which specifies that the MPC lifting slings and MPC lift attachments are designated as important to safety equipment. Contrary to the above, from January 30 to August 3, 2018, the licensee failed to assure that operations of equipment and controls that had been identified as important to safety in the Safety Analysis Report were limited to trained and certified personnel or were under the direct supervision of an individual with training and certification in the operation. Specifically, the licensee's training program:

1. Failed to adequately train and certify the rigger/spotter position involved in the important to safety downloading operation.

2. Failed to have adequate proficiency testing on the controls related to the load indicating device and downloading operations for the VCT operator position.

The team determined that this violation was more than minor because the licensee's failure to establish an adequate training program contributed to the misalignment incident on August 3, 2018. The team assessed and dispositioned this violation in accordance with Section 2.2.2 of the NRC Enforcement Policy. The team characterized the violation as a Severity Level IV violation. The NRC determined the issuance of a Notice is appropriate because the actions to restore compliance have not been fully developed and implemented, and the actions must be effective prior to beginning fuel handling activities. (VIO 07200041/2018-001-03, Failure to establish adequate training program)

The team identified that the simulator canister used for training and dry run demonstrations had a specified outer diameter that was less than that of the actual spent fuel storage canisters being downloaded into the vault. The simulator canister provided approximately 0.75 inch more clearance than the actual canisters loaded with spent fuel. This difference may be acceptable for the dry run activities; however, the difference was not noted in any of the licensee's training materials for rigger/spotters or the VCT operators. This represents a situation of negative training that may have contributed to the August 3, 2018, misalignment incident.

Conclusions

The important to safety lifting equipment and special lifting devices being used for dry cask storage operations met applicable regulatory requirements.

Personnel lacked the proper training, proficiency testing, and certifications to operate important to safety equipment identified in the HI-STORM UMAX SYSTEM FSAR, Revision 4, dated August 14, 2017. This was identified as a violation of 10 CFR 72.190 requirements.

3.1.3 Charter Items 2 and 4

Inspection Scope

"Evaluate the adequacy of the loading procedure(s) with respect to verification of the movement, centering, lowering, and positioning the canister within the ISFSI vault and procedure adherence. Interviews with personnel involved in the ISFSI loading operations should be conducted to evaluate licensee and contractor communications between crane/VCT operators, rigging and spotting staff, cask loading supervisors, radiation protection staff, and licensee oversight personnel. Evaluate the adequacy of pre-job briefings that may have taken place prior to fuel loading operations."

"Based on the review of the procedures and interviews of personnel involved with loading operations, evaluate the adequacy of procedure adherence."

The Special Inspection team reviewed Holtec Procedure HPP-2464-400, "Multi-Purpose Canister Transfer Operations at SONGS," Revision 15; Holtec Procedure HPP-2464-600, "Responding to Abnormal Conditions," Revision 6; SONGS Procedure SO123-0-A7, "Notification and Reporting of Significant Events," Revision 46; and other applicable procedures related to the August 3, 2018, misalignment incident. The team reviewed the pre-job briefing in use by the CLSs. The team discussed ISFSI communications during downloading operations with the licensee and contractor staff.

Observations and Findings

Violation of 10 CFR 72.150, Procedures

The VCT is not equipped with a load-cell to provide the weight of the canister. A hydraulic pressure indication for the lift beam could be used to provide a qualitative means for determining if the slings are not supporting the canister's weight. This pressure indication is displayed on the VCT HMI control panel.

The team identified examples of a violation of 10 CFR 72.150, "Instructions, Procedures, and Drawings." Holtec Procedure HPP-2464-400 provided direction and guidance for verifying canister movement, canister centering operations, and for lowering the canister into the vault. Many steps in the procedure provided direction without quantitative or qualitative means to verify that important to safety steps had been achieved, including detection of a loss-of-load condition and final verification that the canister had been fully downloaded into the vault. For example, step 7.6.12 instructed the VCT operator to continue to raise the VCT lift beam slowly until the full weight of the canister is on the VCT.

However, there is no quantitative direct measurement for the VCT operator to determine when the "full weight" of the canister is indicated on the VCT HMI control panel. The procedure contained a note that the load on the VCT HMI screen may be used to determine if downloader slings had become slack. However the procedure did not direct the VCT operator to monitor the HMI control panel nor provide a qualitative or quantitative value that would notify the VCT operator that the canister had become misaligned and that the VCT was no longer bearing the load of the canister.

Holtec Procedure HPP-2464-400, step 7.6.23, states, if at any time the download slings become slack prior to the canister being in the full down position then immediately stop lowering the canister. During downloading operations there was only one position who could determine whether or not the slings had gone slack. That position was the rigger/spotter who is responsible to monitor the movement of the canister during downloading operations from the elevated JLG basket. The rigger/spotter was observing the slings during the August 3, 2018, downloading evolution. However, the rigger/spotter was only observing the slings for “slack” at the top of the transfer cask.

The procedure did not provide adequate direction to the rigger/spotter to observe the slings near the base of the VCT, which had become slack and were bundling up on the ground. Additionally, the procedure did not provide direction for the rigger/spotter to monitor the height of the canister in relation to the height of the lift beam.

Title 10 CFR 72.150, requires, in part, that the licensee prescribe activities affecting quality by documented instructions or procedures of a type appropriate to the circumstances and must include appropriate quantitative or qualitative acceptance criteria for determining that important activities have been satisfactorily accomplished.

Contrary to the above, from January 30 to August 3, 2018, the licensee failed to prescribe activities affecting quality by documented instructions or procedures of a type appropriate to the circumstances and include appropriate quantitative or qualitative acceptance criteria for determining that important activities have been satisfactorily accomplished. Specifically:

1. Procedure HPP-2464-400, “Multi-Purpose Canister Transfer at SONGS,” Revision 15, step 7.6.23, failed to provide qualitative and quantitative directions for the VCT operator to monitor control panel indications that would identify a canister had become misaligned during downloading operation.

2. Procedure HPP-2464-400, “Multi-Purpose Canister Transfer at SONGS,” Revision 15, step 7.6.23, failed to include adequate instructions for the rigger/spotter to monitor the downloading slings for a slack condition.

The team determined that this violation was more than minor because the licensee’s failure to prescribe adequate procedures contributed to the August 3, 2018, misalignment incident. The team assessed and dispositioned this violation in accordance with Section 2.2.2 of the NRC Enforcement Policy. The team characterized the violation as a Severity Level IV violation. The NRC determined the issuance of a Notice is appropriate because the actions to restore compliance have not been fully developed and implemented, and the actions must be effective prior to beginning fuel handling activities. (VIO 07200041/2018-001-04, Failure to provide adequate instructions of procedures)

Communications

During downloading on August 3, 2018, radiation protection staff directed the CLS and licensee oversight personnel to relocate to a low dose area off of the ISFSI pad. The low dose waiting area was located approximately 150 feet away from the ISFSI operations on the heavy haul path that is approximately 8 feet lower in elevation. From the low dose area, neither the contractor nor licensee oversight staff could observe the

downloading activities. The NRC determined that the removal of oversight staff in an effort to minimize radiation dose without other compensatory measures resulted in inadequate supervisory oversight of important to safety lifting operations.

The communication protocols used by the CLS, VCT operator, and the rigger/spotter was reviewed by the team. The CLS was in direct communications via radio and headsets with the VCT operator and rigger/spotter. The radios provided adequate communication in the noisy environment of the VCT. Communication between the CLS, VCT operator, and the rigger/spotter during the downloading operation was informal. The CLS did not request a reading of the HMI control panel to determine hydraulic pressure and repeat-backs of the location of canister during the downloading process were misunderstood.

Radiation Protection staff were not provided headsets for communications. Radiation Protection staff were able to communicate concerns directly with the CLS, who could communicate radiological concerns to workers, if necessary.

The licensee's oversight personnel were not provided headsets during downloading operations. The licensee did not provide direct oversight of downloading operations. During the August 3, 2018, misalignment incident, neither licensee oversight nor contractor supervision were in a position to directly monitor the downloading operations or the actual condition of the canister.

Conclusions

Dry cask storage procedures did not provide adequate directions for how to determine the downloader slings were slack. The downloading procedure did not include qualitative or quantitative means for determining when a canister had become misaligned. These procedure inadequacies were identified as examples of a violation of 10 CFR 72.150 requirements.

No licensee or contractor oversight personnel were in direct visual observations of the important to safety activities during downloading operations on August 3, 2018. All personnel except the rigger/spotter and VCT operator left the ISFSI pad during downloading operations. Licensee oversight was not a part of the communications between the CLS, the rigger/spotter, and VCT operator during canister downloading operations. Without adequate communications and visual observation, the licensee and the contractor were unable to verify that important to safety dry cask storage activities were adequately performed.

3.1.4 Charter Items 3 and 8

Inspection Scope

"Review and evaluate the licensee's immediate corrective actions taken after the incident for adequacy and notifications to the NRC and safety assessments performed immediately following the incident. Review the licensee's inspection documentation

and/or analysis to determine whether the vault's divider shell experienced any damage that would inhibit the component from performing its designed safety function."

"Investigate the licensee's procedures for reportability to the NRC and determine if the licensee made the correct decision regarding notifications made to the NRC for this incident."

The Special Inspection team reviewed the licensee's initial assessment of the incident through presentations and discussions provided by the licensee. The team reviewed all condition reports and entries made into the licensee's and dry cask storage vendor's corrective action programs regarding the canister misalignment incident, and the condition of the divider shell and canister 29. The team reviewed the notification requirements of 10 CFR 72.75 against the conditions experienced during the August 3, 2018, misalignment incident and reviewed licensee Procedure SO123-0-A7, "Notification and Reporting of Significant Events," Revision 46.

<u>Observations and Findings</u>

Divider Shell Assessment

The licensee immediately stopped all dry cask storage operations following the misalignment incident of August 3, 2018, pending a root cause evaluation to be performed by their dry cask storage vendor, Holtec International. The licensee initiated an apparent cause evaluation to determine if problems in its organization may have contributed to the misalignment incident.

The misalignment incident was entered into the corrective action program by Holtec as FCR 2464-1189. The Holtec FCR was initiated to investigate the August 3, 2018, incident as a human performance issue. This FCR prompted the licensee to initiate Action Request 0818-76588. This action request included an assessment of the condition of the divider shell and canister.

Action Request 0818-76588 described the removal of paint/coating from the divider shell. The action request concluded that the incidental transfer of divider shell coating to the canister shell did not affect the canister's design functions of confinement, shielding, structural, thermal, and criticality. Future actions to address coating presence will be included in the licensee's ISFSI aging management plan. The NRC team reviewed the licensee's assessment for the divider shell and concluded the component can perform its safety functions. Additionally, the licensee's plan to address future inspection of the divider shells in its aging management program was acceptable.

Apparent Violation 10 CFR 72.75, Reporting

The team identified an apparent violation of 10 CFR 72.75 for late notification of 24-hour reporting requirements involving important to safety equipment that was disabled or failed to function as designed when the equipment is required by license condition and no redundant equipment is available and operable to perform the required safety function.

On August 3, 2018, during downloading operations associated with canister 29 the licensee disabled the important to safety slings while downloading a canister (See

Section 2.1 and 2.2). The canister was placed in a potential load drop condition for approximately 45 minutes before the licensee was able to restore the load onto the important to safety slings, thereby restoring the redundant drop protection features.

After the incident, the licensee provided a courtesy notification to the NRC Region IV office at approximately 4 p.m. CDT on the afternoon of August 6, 2018.
Section 10 CFR 72.75(d)(1), would have allowed for notification to be made to the NRC Operations Center as late as 0800 EDT on Monday, August 6, 2018. The courtesy notification made to the regional office did not satisfy the reporting requirements of 10 CFR 72.75. During the August 6, 2018, call, the NRC informed the licensee that a formal report to the NRC was likely required.

Notification of the NRC Operations Center did not occur until the licensee was prompted by the NRC team on September 14, 2018. The condition was reported to the NRC Headquarters Operations Center on September 14, 2018, at 1600 EDT (Event Notification 53605).

Title 10 CFR 72.75(d)(1) requires, in part, that each licensee shall notify the NRC within 24 hours after the discovery of any of the following events involving spent fuel in which important to safety equipment is disabled or fails to function as designed when: (i) the equipment is required by regulation, license condition, or certification of compliance to be available and operable to mitigate the consequences of an accident; and (ii) no redundant equipment was available and operable to perform the required safety function.

Contrary to the above, from August 6 to September 14, 2018, the licensee failed to notify the NRC after discovery of important to safety equipment being disabled and failing to function as designed when required by the Certificate of Compliance to provide redundant drop protection features to prevent and mitigate the consequences of a drop accident and no redundant equipment was available and operable to perform the required safety function.

The licensee's failure to make the required 24-hour notification to the NRC within the required timeframe was identified as an apparent violation of 10 CFR 72.75(d).
(AV 07200041/2018-001-05, Failure to make 24-hour notification)

Conclusions

The licensee concluded that the incidental removal of divider shell coating during downloading operations did not affect the design functions for shielding, structural, and thermal safety functions. The NRC has reviewed the licensee's assessment for the divider shell and has concluded the component can perform its safety functions. Additionally, the licensee's plan to address future inspection of the divider shells in their aging management program is acceptable.

The licensee failed to make the required formal 24-hour NRC notification of the August 3, 2018, event where important to safety equipment was disabled when the equipment was required to mitigate the consequences of an accident and no redundant equipment was available to perform the safety function. This failure was identified as an apparent violation of 10 CFR 72.75(d) requirements.

3.1.5 Charter Item 6

Inspection Scope

"Review the licensee's root cause investigation results, to determine whether the review thoroughly identified all contributing factors and that final corrective actions will be adequate to prevent reoccurrence. Evaluate whether prior operational experience relating to complications or issues associated with canister downloading operations was identified and considered as part of the licensee's root cause investigation and corrective action development."

The Special Inspection team reviewed the causal evaluations that were performed for the August 3, 2018, misalignment incident. Specifically, the team reviewed Holtec International's Root Cause Analysis Report for the canister downloading incident and the licensee's Apparent Cause Evaluation to assess oversight effectiveness during the August 3, 2018, download of canister 29.

Observations and Findings:

Holtec International's Root Cause Evaluation

The licensee directed Holtec to perform a causal evaluation as a follow-up item in condition report action request 0818-76588 that the licensee initiated following the August 3, 2018, misalignment incident. The Holtec causal evaluation identified one root cause and five contributing causes:

- Root Cause: Holtec Management failed to implement appropriate program improvements or the necessary level of oversight commensurate with the complexity and risks associated with downloading operations.
- Contributing Cause 1: Inadequate content in procedures for recognizing special conditions.
- Contributing Cause 2: Design review process did not ensure that unintended consequences of design features were captured.
- Contributing Cause 3: Communication protocols with the chain of command established during canister movement were not well defined.
- Contributing Cause 4: Holtec had not established a continuous learning environment which promoted the use of internal and external operating experience.
- Contributing Cause 5: Holtec Training Program did not fully establish qualification or proficiency requirements for workers performing downloading operations.

Southern California Edison Company's Apparent Cause Evaluation

The licensee initiated an apparent cause evaluation (ACE) to determine how its organization may have contributed to allowing the August 3, 2018, loss-of-load incident to occur. The licensee's apparent causes were related to deficiencies in procedures, training, and in oversight of contractor activities.

- Apparent Cause 1: Management failed to establish a process to ensure that site dry cask storage procedures were technically accurate.
- Apparent Cause 2: Management failed to establish licensee and contractor training to support procedure implementation.
- Apparent Cause 3: Management failed to sufficiently detail contractor Oversight Specialist guidance.
- Contributing Cause 1: ISFSI project management was not routinely observing dry cask storage operations.
- Contributing Cause 2: ISFSI project management was not consistently initiating condition reports for dry cask storage operations that deviated from normal.

Both the licensee and Holtec causal evaluations reviewed many of the items identified by the NRC team. Those items being: procedure adequacy; training adequacy; adequacy of the corrective action program; oversight adequacy; and the inconsistent use of operational experience during routine dry cask storage operations.

The causal evaluations assessed the severity of the canister misalignment incident. The licensee determined that in the event of a canister drop accident from 25 feet into the vault, there was no risk of radioactive exposure to the public. A publicly available version of the licensee's drop analysis summary is available in ADAMS (ADAMS Accession No. ML18330A003). The NRC will continue to review the adequacy of the causal analyses, corrective actions, and potential consequences during a follow-up inspection which is planned to be performed before the resumption of fuel handling activities.

Conclusions

The apparent and root causes for the August 3, 2018, canister misalignment incident involved inadequate training, inadequate procedures, poor utilization of the corrective action program, and insufficient oversight.

3.1.6 Charter Item 7

Inspection Scope

"Review the licensee's planned actions that will address the point loading condition that was experienced by the affected canister. If applicable, review the licensee's analysis that demonstrated the canister will continue to perform as designed for continued storage OR review licensee's inspection plan to safely remove or lift the canister from

the vault to support inspection of the bottom of the canister to demonstrate the canister did not receive any damage that would inhibit the component from continuing to perform as designed."

Observations and Findings

The licensee performed an evaluation to demonstrate the canister continues to meet the design and performance requirements described in the FSAR. The Special Inspection team reviewed the licensee's initial assessment of the canister 29 condition after the misalignment incident.

The preliminary evaluation provided by the licensee stated that both the canister and vault were not expected to have any physical damage that would exceed the pre-defined limits used during receipt inspection and manufacturer acceptance testing. The NRC requested additional analysis to ensure that the canister meets design requirements. Additionally, the licensee is evaluating whether the canister will require increased surveillance frequency for the aging management program. The licensee had not completed the evaluation for NRC review prior to the NRC's inspection exit meeting. This charter item will be reviewed during a future NRC inspection.

Conclusions

The licensee has chosen to provide an analysis to demonstrate that the potential damage to canister 29 during the downloading would meet established acceptance criteria. The NRC determined that additional analysis was required for the NRC to ensure that the canister meets design requirements. This charter item will be reviewed during a future NRC inspection.

3.1.7 Charter Item 9

Inspection Scope

"As directed by regional management, observe resumption of fuel loading operations to verify that corrective actions were effective in addressing deficiencies that contributed to the incident. This should include evaluation of procedure and/or equipment enhancements; review or observation of training and briefings provided to riggers, crane operators, spotters and observers, supervisors and other personnel involved in fuel loading operations."

Observations and Findings

The licensee suspended all fuel handling activities following the August 3, 2018, misalignment incident. The NRC will review the licensee's revised procedures, training plans, equipment modifications, and performance testing (dry runs) of its dry cask storage operations in a future inspection to determine the effectiveness of corrective actions for the incident.

Conclusions

All associated corrective actions for the August 3, 2018, incident had not been completely finalized or implemented by the licensee. The NRC will review the licensee's

revised procedures, training plans, equipment modifications, and performance testing (dry runs) of its dry cask storage operations during a future inspection to determine the effectiveness of corrective actions for the incident.

3.1.8 Charter Item 10

Inspection Scope:

"Determine if the inspection should be elevated to an Augmented Inspection Team (AIT) inspection and promptly notify regional management of any recommendation to escalate the special inspection to an AIT."

As a daily action item, the NRC Special Inspection Team reviewed NRC Inspection Manual Chapter 0309, "Reactive Inspection Decision Basis for Reactors," Enclosure 2, to determine whether any of the facts or details uncovered during the course of the inspection met the deterministic criteria that would require the Special Inspection at SONGS to be elevated to an AIT.

Observations and Findings

The deterministic criteria for an event to be elevated to an AIT effort are delineated in Manual Chapter 0309. The Special Inspection Team did not identify any indication that the August 3, 2018, misalignment incident at SONGS led to a radiological release. Additionally, the incident did not involve the failure of the spent fuel canister, the release of radiological contamination, or external radiation levels that exceeded 10 rads/hr. Consequently, there was no need to elevate the inspection effort to an AIT. The team's daily re-evaluation was communicated to Regional management during the week of onsite inspection effort.

Conclusions

The NRC team did not identify any information that would have required the Special Inspection to be elevated to an AIT effort.

4 Exit Meeting Summary

On September 14, 2018, following the onsite portion of the inspection, the inspectors provided a debrief of the preliminary results to Mr. Tom Palmisano, former Vice President and Chief Nuclear Officer and other members of the licensee staff. The licensee acknowledged the issues presented by the NRC inspection team.

On November 1, 2018, the inspectors presented the final inspection results to Mr. Tom Palmisano, former Vice President and Chief Nuclear Officer and other members of the licensee staff. The licensee acknowledged the issues presented.

On November 8, 2018, the NRC performed a public webinar meeting to discuss the inspection team's preliminary results.

SUPPLEMENTAL INSPECTION INFORMATION

PARTIAL LIST OF PERSONS CONTACTED

Licensee Personnel

A. Bates, Regulatory and Oversight Manager
M. Morgan, Regulatory and Oversight
L. Bosch, Plant Manager
G. Carter, Westinghouse Project Manager
P. Chaudnary, Vice President of Operations, Holtec
J. Manso, ISFSI Sr. Project Manager
T. Palmisano, former Vice President Decommissioning and Chief Nuclear Officer
J. Pugh, Project Engineer
K. Rod, General Manager Decommissioning Oversight
J. Smith, Project Manager, Holtec
M. Soler, Vice President Quality, Holtec

INSPECTION PROCEDURES USED

IP 93812 Special Inspection

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

072-00041/2018-001-01	VIO	Failure to identify and correct conditions adverse to quality (10 CFR 72.172)
072-00041/2018-001-02	AV	Failure to ensure redundant drop protection features were available (10 CFR 72.212)
072-00041/2018-001-03	VIO	Failure to assure that operations of important to safety equipment were limited to trained and certified personnel (10 CFR 72.190)
072-00041/2018-001-04	VIO	Failure to provide adequate instructions or procedures (10 CFR 72.150)
072-00041/2018-001-05	AV	Failure to make 24-hour notification (10 CFR 72.75)

Discussed

None

Closed

None

LIST OF ACRONYMS USED

ADAMS	Agencywide Documents Access and Management System
ADR	Alternative Dispute Resolution
AIT	Augmented Inspection Team
ANSI	American National Standards Institute
AV	Apparent Violation
ASME	American Society of Mechanical Engineers
CFR	*Code of Federal Regulations*
CLS	Cask Loading Supervisor
FCR	Field Condition Report
FSAR	Final Safety Analysis Report
HI-STORM UMAX	Holtec International Storage Module Underground Maximum Capacity
HMI	Human-Machine Interface
IP	Inspection Procedure
ISFSI	Independent Spent Fuel Storage Installation
JLG	Engine or Motor Powered Boom Lifting Device
NOV	Notice of Violation
NRC	U.S. Nuclear Regulatory Commission
MPC	multipurpose canister
PEC	Pre-decisional Enforcement Conference
RIC	Rigger-in-charge
RPT	Radiation Protection Technician
SL	Severity Level
SONGS	San Onofre Nuclear Generating Station
TS	Technical Specification
VCT	Vertical Cask Transporter
VIO	Violation
VVM	Vertical Ventilated Module or vault

INSPECTION CHARTER

TO EVALUATE THE NEAR-MISS LOAD DROP EVENT AT SAN ONOFRE NUCLEAR GENERATING STATION DATED AUGUST 17, 2018 (ML18229A203)

Enclosure 3

UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION IV
1600 EAST LAMAR BOULEVARD
ARLINGTON, TEXAS 76011-4511

August 17, 2018

MEMORANDUM TO: Eric J. Simpson, CHP, Health Physicist
Fuel Cycle and Decommissioning Branch
Division of Nuclear Materials Safety

W. Chris Smith, Reactor Inspector
Engineering Branch 1
Division of Reactor Safety

Marlone X. Davis, Transportation & Storage Safety Inspector
Inspections & Operations Branch
Division of Spent Fuel Management

THROUGH: Janine F. Katanic, PhD, CHP, Chief /RA/ LLH for
Fuel Cycle and Decommissioning Branch
Division of Nuclear Materials Safety

FROM: Troy W. Pruett, Director /RA/
Division of Nuclear Materials Safety

SUBJECT: INSPECTION CHARTER TO EVALUATE THE NEAR-MISS LOAD DROP EVENT AT SAN ONOFRE NUCLEAR GENERATING STATION

A special inspection has been chartered to review the licensee's follow-up investigation, causal evaluation, and planned corrective actions regarding the near-miss drop event involving a loaded spent fuel storage canister at the San Onofre Nuclear Generating Station (SONGS) Independent Spent Fuel Storage Installation (ISFSI) on Friday, August 3, 2018. (License Nos. NPF-10 and NPF-15, Docket Nos. 50-361, 50-362 and 72-41).

CONTACT: Janine F. Katanic, PhD, CHP, FCDB/DNMS
(817) 200-1151

BACKGROUND AND BASIS

On Friday, August 3, 2018, at approximately 1:30 pm (PST), SONGS was engaged in operations involving movement of a loaded spent fuel storage canister into its underground ISFSI storage vault (HI-STORM UMAX storage system). As the loaded spent fuel canister was being lowered into the storage vault using lifting and rigging equipment, the licensee's personnel failed to notice that the canister was misaligned and was not being properly lowered. The licensee continued to lower the rigging and lifting equipment until it believed that the canister had been fully lowered to the bottom of the storage vault. However, a radiation protection technician identified elevated radiation readings that were not consistent with a fully lowered canister. The licensee then identified that the loaded spent fuel canister was hung up on a metal flange near the top of the storage vault, preventing it from being lowered, and that the rigging and lifting equipment was slack and no longer bearing the load of the canister.

In this circumstance, with the important to safety (ITS) rigging and lifting equipment completely down in the lowest position, the ITS equipment was disabled from performing its designed safety function of holding and controlling the loaded canister from a potential canister drop condition. The licensee reported that the canister was resting on a metal flange within the storage vault. It was estimated that the canister could have experienced an approximately 17-18 foot drop into the storage vault if the canister had slipped off the metal flange or if the metal flange failed. This load drop accident is not a condition analyzed in the dry fuel storage system's Final Safety Analysis Report (FSAR).

In response to the discovery that the canister was not fully lowered, the licensee took immediate actions to restore control of the load to the rigging and lifting devices. The estimated time the canister was in an unanalyzed credible drop condition was approximately 45 minutes to 1 hour in duration. The licensee regained control of the load, repositioned the canister, and lowered the canister into the storage vault. The licensee halted all dry fuel storage movement operations in order to fully investigate the incident and develop corrective actions to prevent a recurrence. In addition, the licensee has shared the operational experience with another site with a similar dry fuel storage system.

Region IV became aware of the SONGS "near-miss" incident on Monday, August 6, 2018, when the licensee provided a courtesy notification and described it as a "near-miss" or "near-hit" event. The reporting requirements of the incident are still being evaluated by the Region and discussed with the licensee.

On August 7 and 16, 2018, Region IV and NMSS representatives participated in conference calls with licensee representatives in order to gather additional facts regarding the circumstances of the incident and the licensee's investigation. Region IV is evaluating the information provided by the licensee and is coordinating with the Division of Spent Fuel Management, NMSS.

The NRC is chartering this special inspection pursuant to Management Directive 8.3, "NRC Incident Investigation Program," and NRC Inspection Manual Chapter 0309, "Reactive Inspection Decision Basis for Reactors."

The purpose of the inspection is to investigate the occurrence; interview personnel; observe equipment; and review relevant documentation, including the results of the licensee's investigation and causal analysis, and development and implementation of actions to prevent

recurrence. The licensee has committed to not resume fuel loading operations until after this special inspection and associated reviews are complete. Once the licensee has confirmed its plans to resume fuel loading operations, inspectors will also observe the loading operations to ensure that the corrective actions are adequate. These observations may be conducted as part of this special inspection or as an independent inspection activity, as directed by regional management.

SCOPE

The inspection should seek to address the following items at a minimum:

1. Identify and review all pertinent records, documents, and procedures related to the licensee's downloading operations at the ISFSI pad including but not limited to: worker training and qualifications; rigging equipment qualification, testing, and preventative maintenance; and lifting equipment qualification, testing, and preventative maintenance. Evaluate the adequacy of the above noted procedures, worker training and equipment testing and preparation.

2. Evaluate the adequacy of the loading procedure(s) with respect to verification of MPC movement, centering the MPC over the ISFSI vault, lowering the MPC, and positioning the MPC within the ISFSI vault. Interviews with personnel involved in the ISFSI loading operations should be conducted to evaluate licensee and contractor communications between crane/VCT operators, rigging and spotting staff, cask loading supervisors, radiation protection staff, and licensee oversight personnel. Evaluate the adequacy of pre-job briefings that may have taken place prior to fuel loading operations.

3. Review and evaluate the licensee's immediate corrective actions taken after the event for adequacy of notifications to the licensee and safety assessments performed immediately following the event. Review the licensee's inspection documentation and/or analysis to determine whether the vault's divider shell experienced any damage that would inhibit the component from performing its designed safety function.

4. Based on the review of procedures and interviews of personnel involved with loading operations, evaluate the adequacy of procedure adherence.

5. Interview personnel associated with the event to develop a timeline to ensure the licensee's investigation contained all necessary information to identify all contributing factors and develop adequate corrective actions.

6. Review the licensee's root cause investigation results, to determine whether the review thoroughly identified all contributing factors and that final corrective actions will be adequate to prevent reoccurrence. Evaluate whether prior operational experience relating to complications or issues associated with canister downloading operations was identified and considered as part of the licensee's root cause investigation and corrective action development.

7. Review the licensee's planned actions that will address the point loading condition that was experienced by the affected canister. If applicable, review the licensee's analysis that demonstrated the canister will continue to perform as designed for continued storage OR review licensee's inspection plan to safely remove or lift the canister from the vault to support inspection of the bottom of the canister to demonstrate the canister did not

receive any damage that would inhibit the component from continuing to perform as designed.

8. Investigate the licensee's procedures for reportability to the NRC and determine if the licensee made the correct decision regarding notifications made to the NRC for this event.

9. As directed by regional management, observe resumption of fuel loading operations to verify that corrective actions were effective in addressing deficiencies that contributed to the event. This should include evaluation of procedure and/or equipment enhancements; review or observation of training and briefings provided to riggers, crane operators, spotters and observers, supervisors and other personnel involved in fuel loading operations.

10. Determine if the inspection should be elevated to an AIT and promptly notify regional management of any recommendation to escalate the special inspection to an AIT.

GUIDANCE

The NRC is chartering this special inspection pursuant to Management Directive 8.3, "NRC Incident Investigation Program," and NRC Manual Chapter 0309, "Reactive Inspection Decision Basis for Reactors." The Manual Chapter and Management Directive identify Inspection Procedure 93812, "Special Inspection," for specific use in reviewing events. Planned Dates of Inspection are September 10-14, 2018.

This inspection should emphasize fact-finding in its review of the circumstances surrounding the near-miss canister drop event. Safety concerns identified that are not directly related to near-miss drop event should be reported to NRC management for appropriate action.

Daily briefings with NRC management should occur to discuss the team's progress and preliminary observations.

In accordance with Manual Chapter 0610, a report documenting the results of the inspection should be issued within 30-45 days of the completion of the inspection.

This Charter may be modified should NRC inspectors find significant new information that warrants review. Should you have any questions concerning this charter, please contact Janine F. Katanic at 817-200-1151.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION IV
1600 EAST LAMAR BOULEVARD
ARLINGTON, TEXAS 76011-4511

November 22, 2019

Mr. Doug Bauder
Vice President and Chief Nuclear Officer
Southern California Edison Company
San Onofre Nuclear Generating Station
P.O. Box 128
San Clemente, CA 92674-0128

SUBJECT: SAN ONOFRE NUCLEAR GENERATING STATION INDEPENDENT SPENT FUEL STORAGE INSTALLATION (ISFSI) INSPECTION REPORT 050-00206/2019-003, 050-00361/2019-005, 050-00362/2019-005, 072-00041/2019-001

Dear Mr. Bauder:

This letter refers to the U.S. Nuclear Regulatory Commission's (NRC's) unannounced inspections conducted from July 2019 through September 2019, of the dry cask storage activities associated with your Independent Spent Fuel Storage Installation (ISFSI). The NRC inspectors discussed the results of this inspection with you and other members of your staff during a final telephonic exit meeting conducted on October 21, 2019. The inspection results are documented in the enclosure to this letter.

The inspections examined activities conducted under your license as they relate to safety and compliance with the Commission's rules and regulations and with the conditions of your license. Within these areas, the inspection consisted of selected examination of procedures and representative records, observations of site meetings, performance of independent radiation measurements, and interviews with personnel. Specifically, the inspections reviewed compliance with the requirements specified in the Holtec International HI-STORM UMAX Certificate of Compliance No. 1040 and the associated Technical Specifications, the HI-STORM UMAX Final Safety Analysis Report (FSAR), and Title 10 of the *Code of Federal Regulations* (CFR) Part 72, Part 50, and Part 20.

During the on-site inspections, the NRC observed and confirmed that site personnel completed all required corrective actions identified through causal evaluations for the August 3, 2018, canister misalignment incident to return to fuel loading and transfer operations. Specifically, the NRC inspectors conducted unannounced on-site inspections to evaluate the classroom training, pre-operational training exercises, and a significant number of fuel loading, processing, and dry cask storage transfer evolutions. The NRC inspectors concluded the corrective actions were effectively implemented to ensure the safe transfer of spent fuel to the site's ISFSI.

Based on the results of these inspections, the NRC documented one violation of NRC requirements. The violation was determined to be a Severity Level IV violation of low safety significance under the NRC's traditional enforcement process. The NRC is treating this violation as a non-cited violation (NCV) consistent with Section 2.3.2.a of the NRC Enforcement Policy. If you contest the violation or significance of the NCV, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the U.S. Nuclear

Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-0001, with copies to: (1) the Regional Administrator, Region IV, and (2) the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001.

In accordance with 10 CFR 2.390 of the NRC's "Agency Rules of Practice and Procedure," a copy of this letter, its enclosure, and your response if you choose to provide one, will be made available electronically for public inspection in the NRC Public Document Room or from the NRC's Agencywide Documents Access and Management System (ADAMS). ADAMS is accessible from the NRC's Website at http://www.nrc.gov/reading-rm/adams.html. To the extent possible, your response should not include any personal privacy or proprietary information so that it can be made available to the Public without redaction.

If you have any questions regarding this inspection report, please contact Lee Brookhart at 817-200-1549, or the undersigned at 817-200-1249.

Sincerely,

/RA/

Greg Warnick, Chief
Reactor Inspection Branch
Division of Nuclear Materials Safety

Docket Nos.: 50-206; 50-361; 50-362; 72-041
License Nos.: DPR-13; NPF-10; NPF-15

Enclosure:
Inspection Report 050-00206/2019-003;
050-00361/2019-005; 050-00362/2019-005;
072-00041/2019-001
w/Attachment

cc w/enclosure:
A. Bates, SONGS
L. Bosch, SONGS
W. Matthews, SONGS
G. Perez, CA Dept. of Health
D. Hochschild, CA Energy Commission

U.S. NUCLEAR REGULATORY COMMISSION

REGION IV

Docket Nos.:	50-206; 50-361; 50-362; 72-041
License Nos.:	DPR-13; NPF-10; NPF-15
Report No.:	050-00206/2019-003; 050-00361/2019-005; 050-00362/2019-005; and 072-00041/2019-001
Enterprise Identifier:	I-2019-005-068; I-2019-001-0133
Licensee:	Southern California Edison Company
Facility:	San Onofre Nuclear Generating Station
Location:	San Clemente, CA 92674-012
Inspection Dates:	On-site: July 1-3, 8, 10-11, 15-18, and 22-28; August 12-14, 19-23, and 28; and September 24, 2019
Exit Meeting Date:	October 21, 2019
Inspectors:	Lee Brookhart, Senior ISFSI Inspector Reactor Inspection Branch Division of Nuclear Materials Safety, Region IV Eric Simpson, CHP, Health Physicist Reactor Inspection Branch Division of Nuclear Materials Safety, Region IV W. Chris Smith, Reactor/ISFSI Inspector Reactor Inspection Branch Division of Nuclear Materials Safety, Region IV Marlone Davis, Senior Inspector Inspections and Operations Branch Division of Fuel Management, HQ NMSS
Accompanied by:	Vincent Gaddy, Acting Deputy Director Division of Nuclear Materials Safety, Region IV
Approved By:	Greg Warnick, Chief Reactor Inspection Branch Division of Nuclear Materials Safety, Region IV

Enclosure

EXECUTIVE SUMMARY

NRC Supplemental Inspection Report 050-00206/2019003; 050-00361/2019005; 050-00362/2019005; and 072-00041/2019001

On July 1-3, 8, 10-11, 15-18, and 22-28; August 12-14, 19-23, and 28; and September 24, 2019, the U.S. Nuclear Regulatory Commission (NRC) performed a series of unannounced on-site inspections of dry fuel storage activities of the Independent Spent Fuel Storage Installation (ISFSI) at the decommissioning San Onofre Nuclear Generating Station (SONGS) in San Clemente, California. The on-site inspections were augmented through in-office review of the licensee's condition reports, records, procedures, and other materials gathered and provided prior to and after the on-site portion of the inspections through October 21, 2019. The scope of the inspection was to evaluate and review the licensee's actions related to the resumption of fuel transfer operations from the Unit 2 and Unit 3 spent fuel pools to dry storage following an extended stoppage in loading due to the August 3, 2018, canister misalignment incident. For additional discussions and evaluations of the August 3, 2018, incident, see the NRC Special Inspection Report 050-00206/2018-005, 050-00361/2018-005, 050-00362/2018-005, and 072-00041/2018-001 and Notice of Violation and NRC Supplemental Inspection Report 050-00206/2018-006, 050-00361/2018-006, 050-00362/2018-006, and 072-00041/2018-002 (NRC's Agencywide Documents Access and Management System (ADAMS) Accession Nos. ML18341A172 and ML19190A217, respectively).

During the on-site inspections, the NRC observed and confirmed that the licensee completed all required corrective actions from the licensee's causal evaluations to return to fuel loading operations. Dry cask storage operations were performed in an atmosphere that was free from schedule pressures with an emphasis on procedure adherence. The NRC inspectors verified that the corrective actions implemented were effective to ensure the safe transfer of spent fuel to the site's ISFSI.

<u>Operation of an Independent Spent Fuel Storage Installation, IP 60855</u>

- Several on-site evaluations of the licensee's classroom training lessons and pre-operational dry run training exercises were completed by the inspectors during the inspection period. The inspectors confirmed the corrective actions from the licensee's causal evaluations were adequately implemented regarding licensee oversight, enhanced procedures, and use of new equipment and personnel. The completion of training and dry run exercises of personnel and the demonstration of the newly implemented oversight structure confirmed the licensee was effective in implementing all corrective actions to ensure the safe resumption of fuel loading operations. (Section 1.1)

- The inspectors completed numerous unannounced on-site inspections of the licensee's return to fuel loading operations. The inspections included near 24-hour coverage to evaluate and observe the critical tasks associated with the licensee's spent fuel loading, processing, and downloading operations. The inspectors confirmed the workers were qualified and trained under the licensee's new training program. The procedures utilized in the transfer operations contained the new quantitative and qualitative steps to ensure important tasks were adequately accomplished. During downloading evolutions, the licensee's operations contained the required new personnel, new equipment, and additional oversight to safely place a canister into the UMAX ISFSI. The inspectors determined the licensee was adequately implementing all required corrective actions from the causal

evaluations and the status of the canisters during downloading was constantly monitored and properly handled to avoid any possible misalignment issues. (Section 1.2)

- The inspectors determined that the licensee was placing all relevant identified issues into the Southern California Edison Corrective Action Program (CAP). A large detailed list of issues placed into the site's CAP was reviewed by inspectors and it was determined that the site had established a low threshold of identifying concerns/issues and placing them into the CAP for proper review and resolution. A few of the issues placed into the licensee's CAP, during the inspection period, are further discussed in this report. From all the condition reports reviewed, the inspectors determined the licensee was taking adequate corrective actions to resolve the issues and were appropriately performing reviews for extent of cause and extent of condition when required. (Section 1.3)

Follow-up of Events and Notices of Enforcement Discretion, IP 71153

- The inspectors documented one Severity Level IV, non-cited violation (NCV) related to Licensee Event Report (LER) 2018-002-0 (ADAMS Accession No. ML19050A170), which was previously reviewed and discussed in NRC Supplemental Inspection Report 072-00041/2018-002. Specifically, the violation related to the licensee's failure to conduct past low-profile transporter operations, from January 2018 through August 2018, in accordance with the station's site-specific seismic analysis. The NRC determined that the finding was of low safety significance since the licensee had performed an additional analysis for a revision to the LER 2018-002-1 (ADAMS Accession No. ML19221B590), which bounded any potential contact that could have occurred during a postulated seismic event. The inspectors concluded that the potential impact from adjacent structures (such as light posts) to the conveyance and the loaded transfer cask during a postulated seismic event would not have exceeded any design basis requirements. The licensee had restored compliance prior to resumption of fuel loading activities by revising the transportation procedure to ensure requirements from the site-specific seismic analysis were clearly followed and painted lines along the ISFSI haul path to provide visual boundaries during spent fuel transport operations. The original and revised LERs are closed. (Section 2.2)

Review of 10 CFR 72.48 Evaluations, IP 60857

- Safety screenings had been performed in accordance with the licensee's procedures and 10 CFR 72.48 requirements. All screenings and evaluations reviewed were determined to have been adequately evaluated. (Section 3.3)

REPORT DETAILS

Summary of Plant Activities

The San Onofre Nuclear Generating Station (SONGS) Independent Spent Fuel Storage Installation (ISFSI) consists of two ISFSI designs located adjacent to each other: the Orano Transnuclear (TN) Nuclear Horizontal Modular Storage (NUHOMS) system and the Holtec International Storage Module Underground Maximum Capacity (HI-STORM UMAX) system.

The TN ISFSI contains a total of 63 advanced horizontal storage modules (AHSMs) on the NUHOMS ISFSI pad. Fifty-one of the AHSMs are loaded with the stainless steel dry shielded canisters (DSCs). Spent fuel from all three reactors are stored in 50 of the AHSMs. Greater-than-Class-C (GTCC) waste from the Unit 1 reactor decommissioning project was stored in the 51st module. The twelve empty AHSMs will be available for storage of additional GTCC waste from the decommissioning Units 2 and 3 reactors. The 24PT1-DSCs (Unit 1 fuel) are loaded and maintained under Amendment 0 of Certificate of Compliance (CoC) No. 72-1029 and the 24PT4-DSCs (Units 2 and 3 fuel) are loaded and maintained under Amendment 1 of CoC No. 72-1029. Both CoC amendments were being maintained under NUHOMS Final Safety Analysis Report (FSAR), Revision 5.

The HI-STORM UMAX ISFSI portion was designed to hold 75 Holtec multi-purpose canisters (MPCs). The Holtec MPC-37 canister design can hold 37 pressurized water reactor fuel assemblies in accordance with UMAX CoC No. 72-1040, Amendment 2; HI-STORM UMAX FSAR, Revision 4; and the HI-STORM Flood and Wind (FW) FSAR, Revision 5. The licensee had 35 canisters stored at the UMAX ISFSI at the end of inspection period. Dry cask storage operations had resumed in July 2019, after an 11-month safety stand-down in operations following an August 3, 2018, canister misalignment incident at the UMAX ISFSI.

1 Operation of an Independent Spent Fuel Storage Installation (IP 60855)

1.1 Dry Run and Training Evolutions

a. Inspection Scope

The NRC performed numerous unannounced on-site inspections to evaluate, observe, and assess the licensee's corrective actions following the 11-month stop in fuel movement due to the August 3, 2018, canister misalignment incident.

On July 1-3, 8, and 10-11, 2019, the inspectors observed the licensee successfully complete specific pre-operational dry runs and training of crew personnel on the procedures that were revised based on corrective actions from the licensee's causal evaluations. These dry run demonstrations and training qualifications for crew personnel were targeted on specific evolutions from the revised Procedures HPP-2464-400, "MPC Transfer at SONGS," Revision 22 and HPP-2464-500, "MPC Unloading at SONGS," Revision 8.

The dry run operations observed by the NRC included MPC simulator travel inside the transfer cask on the low-profile transporter from the fuel building along the delineated haul path to the UMAX ISFSI pad, transfer of the transfer cask from the low-profile transporter to the vertical cask transporter (VCT), travel of the VCT up and onto the

ISFSI pad, alignment and securing of the transfer cask to the mating device, and downloading/uploading of the MPC simulator into and out of the UMAX ISFSI vault.

In addition, inspectors periodically observed classroom training while on-site and verified worker qualifications for crews performing dry run training and for workers that had previously completed the exercises.

b. Observations and Findings

The inspectors observed six classroom training lessons under the newly implemented training program. The classroom training was established to qualify the workers to perform fuel transfer operations. The classroom training contained detailed discussions and presentations to ensure all qualified workers understood and were knowledgeable of the changes made in the licensee's transport procedures. The training reviewed the licensee's new oversight structure in place, the new equipment that was required to be used, and the new personnel additions, including their roles and responsibilities. The classroom attendees were attentive, and the classroom environment was interactive with the students actively participating throughout the lessons.

During the observed dry run evolutions, the inspectors were able to observe implementation for several layers of corrective actions as described in the licensee's causal evaluations. Licensee oversight personnel were present at all times during MPC simulator transfer operations as required by the revised oversight task guides. Additional crew personnel and newly implemented canister monitoring equipment (camera, video monitors, and load monitoring devices) were successfully utilized during the training evolutions. The crew followed the new procedures to complete the dry runs. The downloading procedure was revised to include new quantitative and qualitative steps that ensured important tasks were adequately accomplished.

All licensee identified observations, concerns, and issues gathered during the dry run training exercises were captured and placed into the licensee's corrective action program (CAP). The collection of action reports (ARs) were reviewed by the inspectors and were confirmed to be adequately resolved. Resolution of items included additional procedural revisions to enhance the process and additional training on subjects for the crew. The inspectors confirmed the licensee's attention to placing issues into the site's CAP was appropriate and the licensee had established a low-threshold to properly identify, address, and correct issues of concern which could lead to conditions adverse to quality.

c. Conclusions

Several on-site evaluations of the licensee's classroom training lessons and pre-operational dry run training exercises were completed by the inspectors during the inspection period. The inspectors confirmed the corrective actions from the licensee's causal evaluations were adequately implemented regarding licensee oversight, enhanced procedures, use of new equipment and personnel, and placing issues into the licensee's CAP. The completion of training and dry run exercises of personnel and the demonstration of the newly implemented oversight structure confirmed the licensee was effective in implementing all corrective actions to ensure the safe resumption of fuel loading operations.

1.2 Spent Fuel Loading Operations

a. Inspection Scope

This ISFSI inspection included near 24-hour coverage of the loading operations for the critical tasks associated with the licensee's first few canister loading and transfer operations. Inspectors from NRC Region IV office and NRC's Headquarters' Division of Fuel Management observed operations to evaluate and confirm the licensee's corrective actions were being implemented and were effective to ensure safe processing and transport of spent fuel to the site's ISFSI. The inspectors reviewed selected procedures and records to verify ISFSI operations were compliant with Holtec CoC No. 72-1040 license Technical Specifications and the Holtec UMAX FSAR.

On July 15-18, 2019, inspectors observed the transport and downloading of MPC #30 into the SONGS UMAX ISFSI. This canister had been seismically restrained and stored in the Unit 3 fuel building since the August 3, 2018, canister misalignment incident. The NRC inspectors observed operations which included transport of the loaded transfer cask to the UMAX ISFSI and downloading of the canister into the UMAX ISFSI vault.

On July 22-28, 2019, inspectors observed the loading of MPC #31 in the SONGS Unit 2 fuel building. This was the first complete fuel loading operation conducted at SONGS since the August 3, 2018, canister misalignment event. The inspectors observed and evaluated critical evolutions which included 24-hour coverage on fuel movements and fuel verification, heavy lifts associated with the fuel building crane, welding and nondestructive testing of the canister lid-to-shell closure operations, hydrostatic pressure testing, forced helium dehydration, helium backfill, vent/drain port cover welding and nondestructive testing, helium leak testing, radiological surveys, transport of the loaded transfer cask, and downloading of the canister into the UMAX ISFSI vault.

On August 12-14, 19-23, and 28, 2019, inspectors followed-up on the status of site activities and observed the loading of MPC #32. The inspectors provided near 24-hour coverage of the loading operations which included fuel preparation activities, spent fuel movements, heavy lifts associated with the fuel building crane, welding and nondestructive testing of the canister lid-to-shell weld, hydrostatic pressure testing, forced helium dehydration, helium backfill, vent/drain port cover welding and nondestructive testing, helium leak testing, radiological surveys, transport of the loaded transfer cask, and downloading of the canister into the UMAX ISFSI vault.

On September 24, 2019, inspectors reviewed the status of condition reports that had been initiated for recent dry cask storage operations, observed activities related to the processing of MPC #35, and observed the preparations for transport of MPC #34 which was scheduled to be downloaded during the week.

b. Observations and Findings

Throughout the numerous unannounced on-site inspections, the inspectors confirmed that corrective actions associated with the causal evaluations from the August 3, 2018, canister misalignment incident were adequately implemented and effective during the site's return to fuel transfer operations. The inspectors reviewed selected training records to verify work personnel had completed the new training requirements prior to participating in the operations. Through interviews with the crew and loading operation

supervisors, the inspectors confirmed individuals were knowledgeable and competent in their designated roles and responsibilities. The inspectors verified that pre-job briefs and on-the-job-site drills were conducted and contained pertinent information to ensure the crews were prepared to perform and accomplish the critical tasks of each operation.

The inspectors confirmed that revised loading procedures contained the enhanced procedure requirements identified and corrected during the causal evaluation period. These procedure enhancements provided quantitative and qualitative steps to ensure important tasks were adequately accomplished. Additionally, the licensee had performed numerous procedure revisions to capture additional lessons learned from the dry run exercises to enhance the loading operations. The inspectors observed that the dry cask storage operations were performed in an atmosphere that was free from schedule pressures with an emphasis on procedure adherence. During all observed downloading operations, the site adequately implemented all previously identified corrective actions to ensure a misalignment during downloading was avoided or properly resolved.

The inspectors observed and evaluated that the licensee was implementing the new oversight strategies as described in the site's oversight program. The inspectors observed that the licensee implemented multiple layers of oversight participation and provided direct surveillances on all critical activities. The oversight program consisted of three layers of contractor surveillance. The initial layer of licensee oversight was comprised of a contractor technical representative (CTR) which worked alongside each of the vendor's Cask Loading Supervisors (CLS) to perform the work activities. The CTR served in a role of assisting the contractors to meet common goals, facilitate a look ahead perspective, assist with implementation of procedure and safety enhancements, provide coaching, steer evolutions to correct discovered gaps, help resolve issues, and identify areas of improvement.

The second layer of oversight consisted of an oversight specialist (OS). Each shift contained multiple OSs to provide surveillances and assessments on critical activities which were defined in the SCE Procedure G-XV93-PTP, "Pool to Pad Desktop Guide." The OS responsibility includes, evaluating vendor performance for adherence to procedures, capturing issues to place into the corrective action program, providing a holistic approach to ensure conditions different than normal would be recognized, documenting concerns and lessons learned, ensuring the resolution of issues placed in the licensee's CAP, and conducting paired observations and peer observations of other oversight personnel.

The third layer of oversight included personnel from SCE's nuclear quality oversight office which performed independent oversight surveillances and assessments on the vendor's, the CTRs', and the OSs' activities. The inspectors confirmed that each layer of oversight was participating, surveilling, and assessing, as required in the operations that the NRC observed. The inspectors observed that the licensee had established an oversight program that was highly effective in ensuring all activities followed the required procedures, addressed issues of concern and placed issues adverse to quality into the CAP, identified weaknesses and trends that could be improved, and thoroughly took responsibility to ensure operations were performed in a safe and controlled manner.

During the downloading operations, the inspectors confirmed that the operations included the additional trained personnel which were strategically placed in key locations

on man-lifts and on and around the VCT to ensure monitoring of the canister was accomplished as it was lowered into the UMAX vault. The downloading crews utilized the new load monitoring equipment (camera, video monitor, load sensing shackles, wireless headsets, and wireless weight monitoring) to ensure the weight of canister, the position of the canister, and the status of lowering was always known and properly communicated to the crew, supervisors, and oversight. The process to move the spent fuel canister into the UMAX ISFSI was performed in a safe manner over several hours while using the licensee's enhanced procedures and careful observations by the required licensee oversight individuals.

c. Conclusions

The inspectors completed numerous unannounced on-site inspections of the licensee's return to fuel loading operations. The inspections included near 24-hour coverage to evaluate and observe the critical tasks associated with the licensee's spent fuel loading, processing, and downloading operations. The inspectors confirmed the workers were qualified and trained under the licensee's new training program. The procedures utilized in the transfer operations contained the new quantitative and qualitative steps to ensure important tasks were adequately accomplished. During downloading evolutions, the licensee's operations contained the required new personnel, new equipment, and additional oversight to safely place a canister into the UMAX ISFSI. The inspectors determined the licensee was adequately implementing all required corrective actions from the causal evaluations and the status of the canister during downloading was constantly monitored and properly handled to avoid any possible misalignment issue.

1.3 Corrective Actions

a. Inspection Scope

The inspectors reviewed licensee and vendor corrective action reports that had been initiated since the NRC supplemental inspection that concluded in June of 2019. The inspectors reviewed the reports to ensure the issues were being properly addressed, resolved, and the extent of condition or the extent of cause were determined. Additionally, inspectors reviewed the reports to determine if the licensee had addressed causal evaluation corrective actions which included lowering the threshold to place issues into the CAP and that all ISFSI vendor issues on-site would be captured in the SCE CAP.

b. Observations and Findings

The licensee implemented corrective actions to ensure all ISFSI issues with the possibility of being averse to quality were placed into SCE's CAP. The inspectors reviewed a large list of ARs that had been initiated since the last NRC inspection. The detailed list of issues placed into the site's CAP demonstrated the site had established a low threshold of identifying concerns/issues and placing them into the CAP for proper review and resolution. During classroom training lessons, the inspectors verified the new training material included presentations to the crew, supervisors, and oversight to contain a heightened attention for identifying problems and placing them into the SCE CAP.

The inspectors selected ARs for additional review and observed that the conditions described in the ARs were properly addressed, the resolutions were identified to contain the proper corrective actions, and the resolutions addressed extent of condition and extent of cause, when required. Noteworthy condition reports selected by the inspectors for additional follow-up are described below.

Mating Device Closure

One condition report, AR 0719-60949, documented a procedural violation that occurred after the downloading of MPC #30 had been completed on July 18, 2019. During the evolution to remove the Transfer Cask from the stack up configuration, an individual from the cask loading crew inadvertently allowed a portion of their body to cross under the cone of influence of the suspended heavy load (empty transfer cask). The Cask Loading Supervisor (CLS) immediately suspended operations to make the required notifications, initiate condition reports, and discuss the worker's actions with licensee oversight and the crew. During a stand-down meeting to discuss the industrial safety issue with the crew, licensee senior oversight personnel and vendor senior supervisors questioned the CLS if the mating device drawer was in a fully open or partially open position.

Licensee Procedure HPP-2464-400 contained a conservative limit on the time frame that the mating device drawer could be closed. Steps 7.7.8 - 7.7.12 required that the "Mating Device must be removed, or the drawer [fully] opened to establish air cooling [for the MPC] within 4 hours," following downloading and after removing the transfer cask from the stack-up configuration. The CLS was aware of the 4-hour time limitation but had directed the crew to close the drawer approximately half way during the break period for foreign material intrusion concerns. Once it was identified that the drawer was only partially opened, and did not meet the procedure requirement to be fully open, the licensee took immediate corrective actions to fully open the drawer to comply with the procedure.

The licensee estimated that the drawer had been either closed or partially opened for a timeframe of 4 hours and 38 minutes which exceeded the 4-hour time limit in the procedure.

The licensee took immediate corrective actions to place the issue of failing to follow procedure requirements into their corrective action program. In addition, the licensee initiated an event investigation, requested a thermal evaluation from the vendor, and provided training to the cask loading crew and loading oversight staff.

The new thermal analysis from the vendor was initiated to verify the canister did not exceed any design basis limit and evaluate whether the 4-hour procedure time limit for the Mating Device drawer position was appropriate. The UMAX FSAR Section 4.4 and Table 4.1.2 stated the design basis canister (highest thermal canister) would have a peak cladding temperature of 693 °F under normal long-term operations inside the UMAX vault. The canisters loaded at SCE were well below that maximum kW limit (30 kW was the maximum canister loaded at SONGS verses the allowed maximum design basis of approximately 34 kW). The analysis provided by the vendor demonstrated that with the drawer fully closed for 8 hours, the peak cladding temperature, based on the site-specific heat loads, would only rise 27 °F from the steady-state condition. This analysis verified that an 8-hour time limit would still maintain

peak cladding temperatures well below NRC's dry canister peak cladding temperature limits of 752 °F. Additionally, the thermal evaluation confirmed that MPC #30 had never approached nor exceeded any design basis limit from the FSAR.

The inspectors reviewed results of the event investigation, the new thermal analysis, actions taken to train the crew, and the revisions to the procedure. The inspectors confirmed that the partial-closure of the drawer for 38 minutes beyond the conservative time-limit placed in the licensee's procedure, had no effect on the condition of spent fuel and did not cause the canister to exceed any normal operating temperature limit described in the FSAR. Additionally, the licensee's procedure change to extend the time-limit to 8 hours was acceptable and would still preclude a canister from exceeding any established design basis limit described in the FSAR.

The NRC determined this failure to follow a procedure requirement constitutes a minor violation that is not subject to enforcement action in accordance with the NRC's Enforcement Policy. This procedure violation was minor because the limit exceeded was a conservative limit. The FSAR normal operating temperature limits were never approached and the safety significance of exceeding the procedural limit was minor.

Multi-Purpose Canister Storage

During the inspection period, one issue the inspectors closely monitored involved the discovery of rainwater inside some of the unused MPC-37 canisters that were stored at the SONGS site. The two storage areas used at the site included a laydown area inside the owner controlled protected area (PA) and a staging area in Parking Lot #4, adjacent to the site. Both locations were outdoors and subject to ambient environmental conditions.

An SCE oversight individual, while performing routine surveillances of Holtec's activities, initiated AR 0119-19778 in January 2019 which detailed that water had been observed pooling and collecting on the covers of MPCs stored on-site at SONGS, at both storage locations. The licensee determined that the possibility of water intrusion into MPCs being stored at SONGS needed to be addressed and evaluated for impacts to MPC cleanliness and storage requirements.

The likelihood of rainwater intrusion into stored MPC canisters was considered to be low because the MPCs were wrapped in several layers of polymer vapor barrier, had a water tight cap installed, and included a water proof tarpaulin cover that was folded completely over the tops of the stored MPCs. This low likelihood was supported by an initial investigation of an MPC with pooling water on its cover that was inspected in the owner-controlled PA. The inspection showed that no water or corrosion products were present inside of that MPC. Based on those preliminary results, the licensee and Holtec determined that water intrusion was not a problem for any of the other MPCs stored at SONGS.

However, starting in April 2019, the licensee initiated several condition reports that identified that some of the protective barriers and coverings on the MPCs had failed. Specifically, the licensee discovered moisture behind the green vapor barrier sheeting during inspections performed by the licensee. Further inspection identified that the Velcro closures in the MPC tarpaulin covers had failed under the pressure of rainwater pooling on top. The pooling rainwater seeped past the Velcro closures and into the MPC

cavity. The licensee's investigation determined that the protective covers used for the on-site storage of the MPCs failed due to environmental conditions during their prolonged (eleven month) time outdoors. The water identified inside the MPCs had the potential to create adverse conditions, therefore, the condition reports required that an engineering evaluation be performed to determine the effects of moisture intrusion on all MPCs in long term storage and to determine a solution for the problem.

The canisters stored at SONGS were in the possession of Holtec International, the dry cask storage vendor performing pool to pad spent fuel services for the licensee. All unused staged canisters were required to be controlled in accordance with Holtec's Quality Assurance Program. The licensee, SCE, takes possession of the spent fuel canisters as they are placed into service to store the licensee's spent fuel. As such, the licensee performed its inspections of the MPC storage conditions through its contractor oversight role. The licensee worked with Holtec to address the identified issues.

Holtec Procedure HSP-315, "Packaging Shipping Storage of Fabricated and Finished Products," Revision 12, detailed the storage requirements for MPCs. The MPCs at SONGS were to be stored in accordance with the requirements of ANSI 45.2.2, "Packaging, Shipping, Receiving, Storage, and Handling of Items for Nuclear Power Plants," Level C requirements. For Level C storage, Holtec Procedure Step 6.4.3.7.a, stated that items stored in a marine environment shall be stored in a temperature and humidity-controlled building to prevent condensation. However, Step 6.4.3.7.b states that if indoor storage facilities are unavailable, items shall be thoroughly wrapped in a vapor barrier to prevent moisture intrusion.

Holtec issued a condition report (FCR 2464-1417) to address the discovery of degraded MPC protective coverings at SONGS. Holtec issued Response to Request for Technical Information (RRTI) 2464-072 which contained recovery actions that required MPC cleaning and flushing operations. Holtec directed the use of Holtec Procedures HSP-314, "Cleaning of Fabricated Component and Finished Products," Revision 14, and HPP-2464-622, "MPC Cleaning at SONGS," Revision 0, to return the canisters to a Class C cleanliness level as required by ANSI N45.2.1 criteria.

Holtec initiated an MPC flushing, cleaning, and inspection operation to return the MPC canisters to vendor delivered specifications. The canister cleaning operations took place in an open area near the turbine deck so that the turbine deck crane could be used to lift, upend, and manipulate the MPC canisters that required flushing and cleaning. The MPCs were strapped into an Up-Ender, a rig configured to allow the MPC to be fully supported during crane manipulations. The crane allowed the MPC to be positioned onto cribbing that tilted the MPC opening slightly downwards so that the water used in the flushing operations could freely flow into a catch basin. Flushing took place at SONGS using deionized water. The water collected in the catch basin was sampled after the flushing was complete. The water was chemically analyzed to ensure there was no unacceptable chemical contaminates left after the flushing operations.

The inspectors observed the condition of MPCs being stored at SONGS on August 12-14, 2019. At the time of the NRC inspection, Holtec had completed the inspection activities for 14 MPCs of approximately 42 MPCs stored in Parking Lot #4 and in the owner-controlled PA. Ten of the 14 MPCs had standing water at the bottom of the fuel basket. The standing water ranged from 0.5 to 24 inches. The inspectors observed some cleaning operations during its August 19-23, 2019, fuel loading inspection at

SONGS. During these inspections, NRC verified the rainwater and any possible particulates were removed during the flushing operations. The cleaned MPCs were being assessed using ANSI N45.2.1 standards prior to being returned to storage or used in fuel loading operations.

The inspectors concluded that the licensee was performing frequent and thorough surveillances of activities on-site. The licensee was capturing potential issues and adequately resolving those issues through the site's CAP. The actions to clean, flush, analyze, and disposition the canister to meet all applicable requirements prior to use were effective and adequate.

In response to the water intrusion issues, NRC identified a potential issue regarding Holtec's adherence to Holtec Procedure HSP-315, "Packaging, Shipping, and Storage of Fabricated and Finished Products," Revision 12, that occurred at the SONGS site. The inspectors determined that dry cask storage vendor, Holtec, was responsible for MPC storage activities at the site. As such, NRC Region IV has forwarded this issue of concern to the Inspection and Oversight Branch, Division of Fuel Management, for review.

c. Conclusions

The inspectors determined that the licensee was placing all relevant identified issues into the SCE CAP program. A large detailed list of issues placed into the site's CAP was reviewed by inspectors and it was determined that the site had established a low threshold for identifying concerns/issues and placing them into the CAP for proper review and resolution. The inspectors further determined the licensee was taking adequate corrective actions to resolve the issues including a review for extent of cause and extent of condition when required.

2 Follow-up of Events and Notices of Enforcement Discretion (IP 71153)

2.1 Inspection Scope

The inspectors evaluated licensee events to verify the licensee's response and corrective actions were adequate to restore compliance. The inspectors reviewed licensee event reports (LERs) to ensure the reports were timely, accurate, included the required information, and that the required corrective actions had been completed.

2.2 Observations and Findings

a. (Closed) Licensee Event Report 2018-002-0 and 2018-002-1, Spent Nuclear Fuel Transport Conveyance Vehicle Operated Outside Obstacle Clearance Limits

Licensee Event Report 2018-002-0, dated February 14, 2019 (ADAMS Accession No. ML19050A170) was previously discussed in the NRC Supplemental Inspection Report dated July 9, 2019 (ADAMS Accession No. ML19190A217). In the LER, the licensee identified that transporter's center of gravity was not maintained within limitations specified in the site's specific analysis and operations had been conducted too close to adjacent structures (light posts). These past operations were determined to be outside the calculated clearance limits specified in the site's seismic analysis. The licensee identified that the site procedures, at the time, did not provide sufficient detail to

comply with the seismic stability calculation. No actual seismic incidents or collisions with obstacles occurred during past fuel transfer operations and there was no impact to plant personnel or public health and safety. The LER 2018-002-0, at the time, described that an analysis was still in progress to determine if past operations were acceptable.

On August 6, 2019, SCE issued an updated LER 2018-002-1 (ADAMS Accession No. ML19221B590) in accordance with 10 CFR 72.75(d)(1) and (g) for past operations of the low-profile-transporter. The revised LER described that the licensee had performed additional seismic analysis and concluded that there were no safety consequences for traveling along the ISFSI haul route with reduced seismic clearances to adjacent structures and obstacles (light posts). The NRC reviewed the LER revision's additional analysis that bounded any potential contact that could have occurred to the transporter and transfer cask during a postulated seismic event and concluded that the potential impact would not exceed any design basis requirements.

Inspectors found that the updated LER contained adequate content to be closed out in this inspection report. No seismic events have occurred that resulted in damage to the low-profile-transporter or transfer cask during the fuel transfer campaign that began in January 2018. The licensee's failure to follow the site-specific seismic analysis, for past operations, was determined by inspectors to be a violation of NRC requirements (Section 2.2.b). These LERs are closed.

b. Finding Related to the Licensee's Event Report

The licensee's event notification EN #53798 documented that past low-profile transporter (HI-PORT) operations had not been conducted within the requirements of the original seismic evaluation HI-2167363, "Seismic Stability Analysis of HI-TRAC on HI-PORT at SONGS," Revision 4. At times the low-profile-transporter was operated outside of the 10-inch travel height restriction of the seismic analysis. Also, the proximity distance to adjacent structures was never formally outlined along the haul path from the fuel buildings to the ISFSI pad. Evaluation HI-2167363, Section 5.0, "Conclusions," stated that the loaded HI-PORT height should be setup to have a maximum clearance of 10 inches between the dropdeck and haul path and a minimum clearance of 32 inches should be maintained between the outer edges of the HI-PORT and adjacent safety related structures or structures that may adversely affect the HI-PORT along the haul path at all times.

Title 10 CFR 72.212(b)(3), requires, in part, that the general licensee shall ensure that each cask used conforms to the terms, conditions, and specifications of a Certificate of Compliance as listed in 10 CFR 72.214.

Title 10 CFR 72.214 states, in part, that Certificate Number 1040 [Docket Number 072-01040] Amendment Number 2, effective date January 9, 2017, is an approved cask for storage of spent fuel under the conditions specified in the Certificate of Compliance for the Holtec HI-STORM UMAX Storage System.

Certificate of Compliance 072-01040, Appendix B, Technical Specification 3.4.15 requires, in part, the loaded transfer cask and its conveyance shall be evaluated to ensure, under the site-specific Design Basis Earthquake (DBE), that the cask and its conveyance does not tip-over or slide off the haul route.

Contrary to the above, from January 30, 2018, to August 3, 2018, the licensee failed to ensure the cask and its conveyance was evaluated under the site-specific DBE. Specifically, the licensee identified that past HI-PORT transportation operations were not evaluated under the site-specific DBE, since operations were conducted outside the height and stand-off requirements in seismic evaluation HI-2167363.

This violation was dispositioned per the traditional enforcement process using Section 2.3 of the NRC's Enforcement Policy. The NRC determined that the finding was of low safety significance since the licensee had re-performed the evaluation, addressed the deviation that occurred, and demonstrated the canister and its conveyance would not have tipped over, slipped off, or experienced damage beyond design basis requirements along the haul route during those transportation operations due to being operated outside of the identified limits. This finding was determined by inspectors to be of more than minor safety significance, since if left uncorrected, the deficiency could lead to a more significant safety concern.

Consistent with the guidance in Section 1.2.6.D of the NRC Enforcement Manual, if a violation does not fit an example in the Enforcement Policy Violation Examples, it should be assigned a severity level: (1) commensurate with its safety significance; and (2) informed by similar violations addressed in the Violation Examples. The violation was evaluated to be similar to a Severity Level IV violation in Enforcement Policy Section 6.1.d.1.

The licensee entered the finding into its CAP as AR 1218-46759. The licensee restored compliance, before fuel loading resumption activities, by (1) changing the transportation procedure to ensure requirements from the site-specific seismic analysis were clearly outlined and followed, (2) painting lines along the ISFSI haul path to provide visual boundaries that the HI-PORT operator should not cross during spent fuel transport operations, and (3) revising the site-specific seismic analyses to bound transportation operations conducted at the site. Because the licensee entered the issue into its CAP, the safety significance of the issue was low, and the issue was not repetitive or willful, this Severity Level IV violation was treated as an NCV, consistent with Section 2.3.2.a of the Enforcement Policy (NCV 07200044/2019-001-01; "Failure to Ensure the Loaded Transfer Cask and its Conveyance was Evaluated Under the Site-Specific DBE" (10 CFR 72.212(b)(3)).

2.3 <u>Conclusions</u>

The inspectors documented one Severity Level IV, non-cited violation (NCV) related to the licensee's failure to conduct past transportation operations, utilizing the low-profile transporter, in accordance with the station's site-specific seismic analysis. The NRC determined that the finding was of low safety significance since the licensee had performed an additional analysis for a revision to the LER which bounded any potential contact that could have occurred during a postulated seismic event. The inspectors concluded that the potential impact to the transfer cask would not have exceeded any design basis requirements. The licensee restored compliance, prior to fuel loading resumption activities, by changing the transportation procedure to ensure requirements from the site-specific seismic analysis were clearly outlined and followed and revised the site-specific seismic analyses to bound transportation operations conducted at the site. The original and revised LERs are closed.

3 Review of 10 CFR 72.48 Evaluations (IP 60857)

3.1 Inspection Scope

The licensee's 10 CFR 72.48 screenings and evaluations performed since the NRC's last ISFSI inspection (ADAMS Accession No. ML19190A217) were reviewed to determine compliance with regulatory requirements.

3.2 Observations and Findings

The licensee's 10 CFR 72.48 screenings and evaluations for ISFSI program changes since June 2019 were reviewed to determine regulatory compliance. The licensee had performed a number of procedure revisions and some equipment or process changes under the 72.48 process since the last inspection. NRC inspectors reviewed the 72.48 screens for those procedure changes and design change packages made within the ISFSI program. None of the screens led to a full 10 CFR 72.48 safety evaluation. All screenings were determined to be adequately evaluated.

3.3 Conclusions

All required safety screenings and safety evaluations had been performed in accordance with procedures and requirements of 10 CFR 72.48. All screenings and safety evaluations reviewed were determined to have been adequately evaluated.

4 Exit Meeting Summary

On October 21, 2019, the NRC inspectors presented the final inspection results to Mr. Doug Bauder, Vice-president and Chief Nuclear Officer, Southern California Edison and other members of the licensee's staff. The licensee acknowledged the issues presented.

SUPPLEMENTAL INSPECTION INFORMATION

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INSPECTION PROCEDURES USED

IP 60855	Operation of an Independent Spent Fuel Storage Installation
IP 71153	Follow-up of Events and Notices of Enforcement Discretion
IP 60857	Review of 10 CFR 72.48 Evaluations

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened and Closed

07200044/2019-001-01	NCV	Failure to Ensure the Loaded Transfer Cask and its Conveyance was Evaluated Under the Site-specific DBE

Closed

2018-002-0	LER	Spent Nuclear Fuel Transport Conveyance Vehicle Operated Outside Obstacle Clearance Limit, Revision 0
2018-002-1	LER	Spent Nuclear Fuel Transport Conveyance Vehicle Operated Outside Obstacle Clearance Limit, Revision 1

LIST OF ACRONYMS USED

ADAMS	Agencywide Documents Access and Management System
AHSM	Advanced Horizontal Storage Module
ANSI	American National Standards Institute
AR	SCE Action Request
CAP	Corrective Action Program
CFR	*Code of Federal Regulations*
CLS	Holtec Cask Loading Supervisor
CoC	Certificate of Compliance
CTR	SCE Contractor Technical Representative
DBE	Design Basis Earthquake
DSC	Dry Shielded Canister
EN	Event Notification
FCR	Holtec Field Condition Report
FSAR	Final Safety Analysis Report
GTCC	Greater than Class C
HI-STORM FW	Holtec International Storage Module Underground Flood and Wind
HI-STORM UMAX	Holtec International Storage Module Underground Maximum Capacity
IP	Inspection Procedure
ISFSI	Independent Spent Fuel Storage Installation
LER	Licensee Event Report
NCV	Non-Cited Violation
NRC	U.S. Nuclear Regulatory Commission
NUHOMS	Nuclear Horizontal Modular Storage
MPC	multipurpose canister
OS	SCE Oversight Specialist
PA	Protected Area
RRTI	Response to Request for Technical Information
SCE	Southern California Edison
SONGS	San Onofre Nuclear Generating Station
TN	Orano Transnuclear
VCT	Vertical Cask Transporter

SAN ONOFRE NUCLEAR GENERATING STATION INDEPENDENT SPENT FUEL STORAGE INSTALLATION (ISFSI) INSPECTION REPORT 050-00206/2019-003, 050-00361/2019-005, 050-00362/2019-005, 072-00041/2019-001 DATED – NOVEMBER 22, 2019

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Division of Spent Fuel Storage and Transportation
Interim Staff Guidance - 1, Revision 2
Classifying the Condition of Spent Nuclear Fuel for Interim Storage and Transportation Based on Function[1]

Issue

This Interim Staff Guidance (ISG) provides guidance to the staff on classifying spent nuclear fuel as either (1) damaged, (2) undamaged, or (3) intact, before interim storage or transportation. This is not a regulation or requirement and can be modified or superseded by an applicant with supportable technical arguments.

Regulatory Basis

Fuel-Specific Regulations:

A fuel-specific regulation means a characteristic or performance requirement of the fuel specifically named in the applicable Code of Federal Regulations (CFR). These are regulations that specify capabilities that the spent nuclear fuel (SNF) must have. Examples include:

10 CFR 71.55(d) states, in part: "A package used for the shipment of fissile material must be so designed and constructed and its contents so limited that under the tests specified in 10 CFR 71.71 ('Normal conditions of transport')

(1) The contents would be subcritical.
(2) The geometric form of the package contents would not be substantially altered."

10 CFR 72.44(c) states, in part: "Technical specifications must include requirements in the following categories: (1) Functional and operating limits . . . (I) . . . for an ISFSI or MRS are limits on fuel or waste handling and storage conditions that are found to be necessary to protect the integrity of the stored fuel or waste container, to protect employees against occupational exposures and to guard against the uncontrolled release of radioactive materials . . . "

10 CFR 72.122(h)(1) states: "The spent fuel cladding must be protected during storage against degradation that leads to gross ruptures or the fuel must be otherwise confined such that degradation of the fuel during storage will not pose operational safety problems with respect to its removal from storage. This may be accomplished by canning of consolidated fuel rods or unconsolidated assemblies or other means as appropriate."

10 CFR 72.122(l) states: "Retrievability. Storage systems must be designed to allow ready retrieval of spent fuel, . . . for further processing or disposal."

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[1]Formerly entitled "Damaged Fuel."

System-Related Regulations:

A transportation and storage system must satisfy all applicable regulations in 10 CFR Parts 71 or 72. A system-related regulation is a performance requirement placed on the fuel for the system (i.e., transportation or storage cask) to meet a regulation that does not specifically require performance capabilities of the SNF. Examples include:

10 CFR 71.55(e) states in part: "A package used for the shipment of fissile material must be so designed and constructed and its contents so limited that under the tests specified in 10 CFR 71.73 ('Hypothetical accident conditions'), the package would be subcritical."

Note: This regulation does not place a specific requirement on the SNF. However, if the package requires the SNF to maintain its geometric configuration to ensure subcriticality, then a function is imposed on the SNF.

10 CFR 72.122(h)(5) states: "The high level radioactive waste and reactor related GTCC waste must be packaged in a manner that allows handling and retrievability without the release of radioactive materials to the environment or radiation exposures in excess of Part 20 limits . . . "

10 CFR 72.124(a) states: "Design for criticality safety. Spent fuel handling, packaging, transfer, and storage systems must be designed to be maintained subcritical and to ensure that, before a nuclear criticality accident is possible, at least two unlikely, independent, and concurrent or sequential changes have occurred in the conditions essential to nuclear criticality safety. The design of handling, packaging, transfer, and storage systems must include margins of safety for the nuclear criticality parameters that are commensurate with the uncertainties in the data and methods used in calculations and demonstrate safety for the handling, packaging, transfer, and storage conditions and in the nature of the immediate environment, under accident conditions."

Note: If the SNF must have certain characteristics or behave in a specified manner to maintain the required margins, a function is placed on the SNF.

10 CFR 72.128 states in part: "Spent fuel storage . . . must be designed to ensure adequate safety under normal and accident conditions. These systems must be designed with (2) suitable shielding for radioactive protection under normal and accident conditions, (3) confinement structures and systems . . . "

Note: If proper functioning of the shielding or containment requires that the SNF maintain its configuration, then a function is placed on the SNF.

Applicability

This guidance applies to reviews of dry cask storage systems and transportation casks conducted in accordance with NUREG-1536, "Standard Review Plan for Dry Cask Storage Systems" (January 1997); NUREG-1567, "Standard Review Plan for Spent Fuel Dry Storage Facilities" (March 2000); or NUREG-1617, "Standard Review Plan for Transportation Packages for Spent Nuclear Fuel" (March 2000). This revision of ISG-1 supersedes any definitions of damaged, grossly damaged, or intact fuel in the above Standard Review Plans.

This revision supersedes ISG-1, Revision 1, "Damaged Fuel," in its entirety, and is applicable to both as-built and reconstituted fuel assemblies.

Definitions

1. Spent Nuclear Fuel (SNF) - See 10 CFR Part 72.3 for definition. This term has been used in the nuclear industry, at different times, to mean the fuel pellets, the rod, or entire fuel assembly. Unless specifically modified, the term will refer to both the rods and fuel assembly.

2. Damaged SNF - Any fuel rod or fuel assembly that cannot fulfill its fuel-specific or system-related functions.

3. Undamaged SNF - SNF that can meet all fuel-specific and system-related functions. As shown in Figure 1, undamaged fuel may be breached. Fuel assembly classified as undamaged SNF may have "assembly defects."

4. Breached spent fuel rod - Spent fuel rod with cladding defects that permit the release of gas from the interior of the fuel rod. A breached spent fuel rod may also have cladding defects sufficient to permit the release of fuel particulate. A breach may be limited to a pinhole leak or hairline crack, or may be a gross breach.

5. Pinhole leaks or hairline cracks - Minor cladding defects that will not permit significant release of particulate matter from the spent fuel rod, and therefore present a minimal as-low-as-is-reasonably-achievable concern, during fuel handling and retrieval operations. (See discussion of gross defects for size concerns.)

6. Grossly breached spent fuel rod - A subset of breached rods. A breach in spent fuel cladding that is larger than a pinhole leak or a hairline crack. An acceptable examination for a gross breach is a visual examination that has the capability to determine the fuel pellet surface may be seen through the breached portion of the cladding. Alternatively, review of reactor operating records may provide evidence of the presence of heavy metal isotopes indicating that a fuel rod is grossly breached. (See discussion for size concerns.)

7. Intact SNF - Any fuel that can fulfill all fuel-specific and system-related functions, and that is not breached. Note that all intact SNF is undamaged, but not all undamaged fuel is intact, since under most situations, breached spent fuel rods that are not grossly breached will be considered undamaged.

8. Can for Damaged Fuel - A metal enclosure that is sized to confine one damaged spent fuel assembly. A fuel can for damaged spent fuel with damaged spent-fuel assembly contents must satisfy fuel-specific and system-related functions for undamaged SNF required by the applicable regulations.

9. Assembly Defect - Any change in the physical as-built condition of the assembly with the exception of normal in-reactor changes such as elongation from irradiation growth or assembly bow. Examples of assembly defects: (a) missing rods; (b) broken or missing grids or grid straps (spacers); and (c) missing or broken grid springs, etc. An assembly with a defect is damaged only if it can't meet its fuel-specific and system-related functions required by the applicable regulations.

Note: See Appendix for default definition of damaged SNF.

Background

Damaged Fuel

Previous definitions of damaged fuel have identified specific characteristics of the fuel that classify it as damaged, irrespective of whether the fuel is being stored or transported and independent of the design of the storage or transportation system. In this guidance, damaged fuel is defined in terms of the characteristics needed to perform the fuel-specific and system-related functions. Thus, the characteristics of damaged spent fuel may depend on (1) whether the fuel is being stored or transported, and (2) the design of the storage or transportation system.

The materials properties, and possibly the physical condition, of a fuel rod or assembly can be altered during irradiation, storage, or transportation. If this alteration is large enough to prevent the fuel or assembly from performing its fuel-specific or system-related functions during storage, transportation, or both then the fuel assembly is considered damaged.

To determine whether a fuel assembly is undamaged, the following should be delineated:

1) Whether the definition is applicable to storage, transportation or both;

2) The functions the applicant has imposed on the fuel rods and assembly by either fuel-specific or system-related functions to meet a regulatory requirement for the designated phase (storage, transportation, or both);

3) The mechanisms of change (alteration mechanisms) or the characteristics of the fuel that could potentially cause the fuel to fail to meet its fuel-specific or system-related functions;

4) An acceptable analysis showing that the fuel with the designated characteristics will meet the fuel-specific and system-related functions when the mechanisms considered in item #3, above, are evaluated; and

5) The physical characteristics of the fuel, based on item #4, above, that could cause the fuel or assembly to be classified as "damaged."

The "Discussion" section illustrates this methodology in the example.

Damaged SNF, as defined in this guidance, will only be approved for the activity (storage, transportation, or both) for which the application is being submitted. Note that the "default" definition of damaged SNF, derived from ANSI N14.33-2005, is provided in the appendix of this

guidance for those that do not want to perform the assessment outlined in item numbers 1 through 5 above [2]. The default definition, however, may not take full advantage of the flexibility of the performance-based definition of damaged fuel provided in this guidance. This default definition may be more restrictive than necessary, depending on the design of the storage or transportation cask. For example, the default definition of damaged SNF indicates that SNF must be classified as damaged if an individual fuel rod is missing from an assembly. However, if an analysis shows that all fuel-specific and system-related functions will be met (e.g., subcriticality will be maintained, that the SNF assembly will be retrievable and that the structural properties of the assembly are not compromised by the missing rod) the assembly may be classified as undamaged, per this ISG.

Discussion

The performance-based definition [3,4] of damaged SNF provided in this ISG minimizes the quantity of damaged fuel requiring alternative handling paths, while still addressing applicable system-related regulations concerning criticality control, thermal limitations, structural integrity, confinement, and shielding.

A. Grossly Breached SNF Cladding

The regulations in 10 CFR 72.122(h) state “The spent fuel cladding must be protected during storage against degradation that leads to gross ruptures or the fuel must be otherwise confined such that degradation of the fuel during storage will not pose operational safety problems with respect to its removal from storage.” However, there is no such requirement in 10 CFR 71. Hence, grossly breached fuel is always considered damaged for storage, but may, or may not, be considered damaged for the purposes of transportation depending on whether other regulations, such as criticality, can be met.

In dry cask storage and transportation systems, a gross cladding breach should be considered as any cladding breach that could lead to the release of fuel particulate greater than the average size fuel fragment. A pellet is ~1.1 centimeters in diameter in 15 x 15 Pressurized-Water Reactor (PWR) assemblies. Pellets from a Boiling-Water Reactor (BWR) are somewhat larger, and those from 17 x 17 PWR assemblies are somewhat smaller. The pellet's length is slightly longer than its diameter. During the first cycle of irradiation in-reactor, the pellet fragments into 25-35 smaller interlocked pieces, plus a small amount of finer powder, due to, pellet-to-pellet abrasion. When the rod breaches, about 0.1 gram of this fine powder may be carried out of the fuel rod at the breach site [5]. Modeling the fragments as either spherical- or pie-shaped pieces indicates that a cladding-crack width of at least 2-3 millimeters would be required to release a fragment. Hence, gross breaches should be considered to be any cladding breach greater than 1 millimeter.

A review of reactor operating records, ultrasonic testing, and sipping (if done in a timely fashion) can be used to classify rods as unbreached or, breached. Evidence of only gaseous or volatile decay products (no heavy metals) in the reactor coolant system is accepted as evidence that a cladding breach is no larger than a pinhole leak or hairline crack. Records that show the presence of heavy metal isotopes that are characteristic of fuel release in the reactor coolant system indicate gross breaches in the cladding. Likewise, visual examination may also be used to determine if a cladding breach is gross, if the breached rod can be positively identified.

Because cladding openings larger than 1 millimeter should expose the fuel pellet to visual sighting, visual examination of the breached rod can be used to determine if a breach is gross. However, visual examination is not an acceptable method of confirming intact (undamaged) fuel for assemblies that have indicated leakage.

It should be noted; however, that undamaged spent-fuel rods with pinhole leaks and/or hairline cracks will expose the fuel pellets to the canister or cask atmosphere. If that atmosphere is oxidizing, then the fuel pellet may oxidize and expand, placing stress on the cladding. The expansion may eventually cause a large split in the cladding, resulting in spent fuel that must be classified as damaged (for storage and possibly also for transportation) due to gross breaches in the cladding. Since fuel oxidation and cladding splitting follow Arrhenius time-at-temperature behavior, fuel rods with pinholes or hairline cracks that are exposed to an oxidizing atmosphere may experience this type of additional cladding damage. ISG-22 “Potential Rod Splitting Due to Exposure to an Oxidizing Atmosphere During Short-Term Cask Loading Operations in LWR or other Uranium Oxide Based Fuels” [6] provides information regarding prevention of this phenomenon. Before handling undamaged rods with pinhole leaks and/or hairline cracks in an oxidizing atmosphere, the potential fuel and cladding degradation at the temperature of interest for the duration of the process should be assessed.

B. Fuel Assembly with Defects

Damage under this guidance refers to alterations of the fuel assembly that prevent it from fulfilling its fuel-specific or system-related functions. Defects such as dents in rods, bent or missing structural members, small cracks in structural members, missing rods, etc., need not be considered damaged if the applicant can show that the fuel assembly with these defects still fulfils its fuel-specific and system-related functions. This may be done using calculations based on approved codes, situation-specific data, or reasoned engineering arguments.

C. Canning Damaged Fuel

Spent fuel that has been classified as damaged for storage must be placed in a can designed for damaged fuel, or in an acceptable alternative. The purpose of a can designed for damaged fuel is to (1) confine gross fuel particles, debris, or damaged assemblies to a known volume within the cask; (2) to demonstrate that compliance with the criticality, shielding, thermal, and structural requirements are met; and (3) permit normal handling and retrieval from the cask. The can designed for damaged fuel may need to contain neutron-absorbing materials, if results of the criticality safety analysis depend on the neutron absorber to meet the requirements of 10 CFR 72.124(a).

D. Relationship of Spent Fuel Populations

The applicant will designate the population of spent fuel for which the cask system was designed (e.g., type of fuel, minimum cooling time, burnup limitations, arrays, manufacturers, cladding types, etc.) This population may contain breached rods. Some of these breached rods may be grossly breached. It may also contain assemblies with defects, such as missing rods, missing grid spacers, or damaged spacers. The populations of breached rods, grossly breached rods, and assemblies with defects are determined by in-reactor behavior and ex-reactor handling.

Each of these populations must be classified as damaged or undamaged after the storage or transportation system has been designated. For example, an applicant might propose the use of air as a cover gas in its design of a storage cask. The applicant might also propose this cask for use in storing spent fuel with cladding breaches that are hairline cracks or pinhole leaks. However, if the spent fuel in the cask will operate at a sufficiently high temperature for a long enough time, then oxidation of fuel pellets in breached rods could occur resulting in gross breaches. If this is the case, the breached spent fuel should be considered damaged because grossly breached rods do not meet the requirements of 10 CFR 72.122(h)(1). Also, in this case because the geometric form of the package contents could be substantially altered, the spent fuel would also be classified as damaged for transportation because the requirements of 10 CFR 71.55(d)(2) might not be met. If an inert atmosphere was used instead of air, only grossly breached rods would be considered damaged for storage. This concept is illustrated in Figure 1, "Relationship of Spent Fuel Populations."

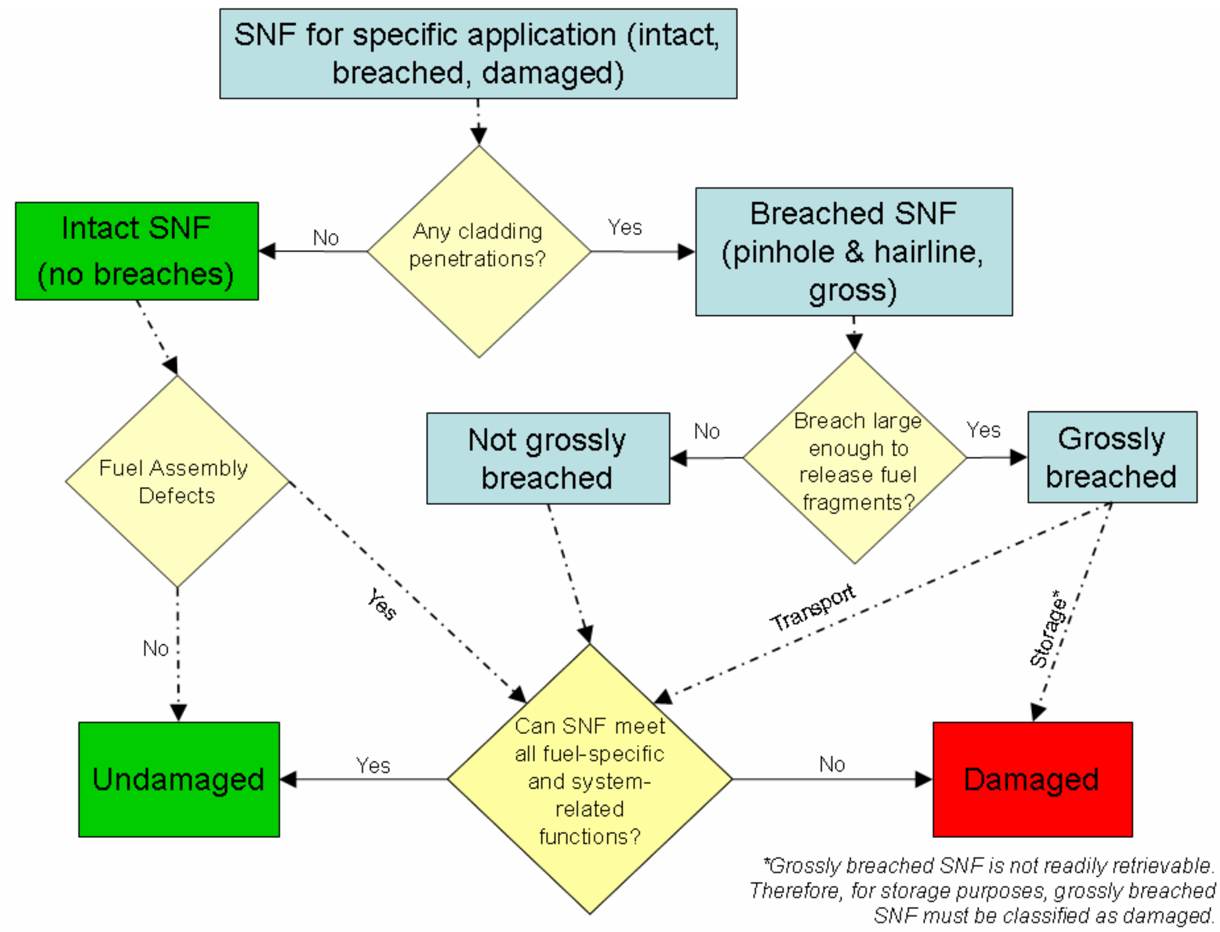

Figure 1 - Relationship of Spent Fuel Populations

E. Example of Methodology

The following example is given to illustrate the general methodology adopted in this ISG. This is only an example of the methodology and should not be construed as approved characterization of damaged fuel.

Example:

Situation - The vendor of a dual-purpose cask wants to store and transport low-burnup PWR fuel in an inert atmosphere and within the temperature limits recommended in ISG-11, Revision 3, “Cladding Considerations for the Transportation and Storage of Spent Fuel” [7]. The vendor wants to store assemblies having rods with breaches containing only pinholes or hairline cracks, and assemblies having one or more outer grid straps with defects at three or more grid locations without canning them. The vendor is only applying for a storage license at this time but wants to be reasonably certain that the fuel will also be transportable.

Activity - Storage and transportation

Fuel-specific or system-related functions imposed on rods and assemblies - 10 CFR 72.122(h)(1), regarding gross ruptures, and 10 CFR 72.122(l), concerning retrievability, must be met for storage. 10 CFR 71.55(d), requiring the system to remain subcritical and unchanged during normal transport, must be met. The vendor believes that all the remaining system requirements, except for the subcriticality requirement, can be met, without imposing any limitations on the fuel, if the fuel is within the bounds stated in the situation.

Mechanisms - There are no mechanisms for the pinhole leaks and hairline cracks to evolve into gross breaches since the atmosphere is inert and the temperature is controlled. To be retrievable, the assemblies with missing grid straps must be able to withstand design basis events in a storage cask. Since the applicant also wants these assemblies to be considered undamaged for transportation, the behavior of the assemblies under both normal and hypothetical accident transportation conditions in 10 CFR 71 must be evaluated. For example, for normal transportation conditions, the applicant must show that the assemblies with the most missing grid straps in the worst locations can withstand both normal vibration and a one-foot drop and remain in their original physical configuration. Additionally, for hypothetical accident conditions, the analysis must indicate, among other things, that the system will meet shielding and subcriticality requirements when placed under the mechanical and thermal loads specified in 10 CFR 71.

Analysis - The applicant conducts an analysis to satisfactorily demonstrate that the assembly with three missing grid straps in the worst configuration remains intact for 1) normal transportation conditions; 2) cask tip-over; and 3) regulatory accident conditions. Further acceptable analysis indicates that all the system-related regulations are met, if the fuel with the characteristic limitations (as noted in Characteristics section below), stays structurally intact.

Characteristics - Assemblies containing breached rods with up to three grid straps missing will be considered undamaged for the purposes of storage. Analysis shows that these assemblies could probably also be considered undamaged for transportation, but fuel with these characteristics will be evaluated and approved as part of a later application for the transportation cask certification.

Records

Records documenting the classification of spent fuel shall comply with the provisions of 10 CFR 72.174, “Quality Assurance Records”; 10 CFR 72.72, “Material Balance, Inventory, and Records Requirements for Stored Material”; 10 CFR 71.91, “Records”; and 10 CFR 71.135, “Quality Assurance Records.” Inspection records will be maintained for the lifetime of the container.

Quality Assurance

Activities related to inspection, evaluation, and documentation of damaged spent fuel for dry storage shall be performed in accordance with a quality assurance program, as required in 10 CFR Part 72, Subpart G, “Quality Assurance.” Activities related to inspection, evaluation, and documentation of damaged spent fuel for transport shall be performed in accordance with a quality assurance program, as required in 10 CFR Part 71, Subpart H, “Quality Assurance.”

Recommendations

The staff recommends that: (1) the definitions in NUREG-1536, NUREG-1567, NUREG-1617, ISG-8, Revision 2 and ISG-11, Revision 3 be revised to incorporate the definitions listed above; and (2) the appropriate chapters of each NUREG be revised to include the discussion section of this ISG.

In addition, the suggestion in NUREG-1617 (canning damaged fuel is necessary for the purposes of transportation) should be modified to be consistent with this ISG, nothing that canning damaged fuel may, in some cases, be necessary to meet the requirements of 10 CFR Part 71.

The words “intact fuel,” in the Applicability section of Revision 2 of ISG-8, “Burnup Credit in the Criticality Safety Analyses of PWR Spent Fuel in Transport and Storage Casks,” should be replaced with “undamaged fuel.” “Intact commercial spent fuel" in the last paragraph of the “Issue” section of Revision 3 of ISG-11, “Cladding Considerations for the Transportation and Storage of Spent Fuel,” should read “undamaged commercial spent fuel.” The first sentence of the Issue Section of ISG-22 should be modified to be consistent with the current definitions. “Rev 1,” should be changed to “Rev 2,” with the new title to ISG-1 Rev 2 and “intact fuel” should be changed to “unbreached fuel.”

Approved :
/RA/ | May 11, 2007
E. William Brach, Director | Date
Division of Spent Fuel Storage
and Transportation

Attachment: Appendix

APPENDIX

Default Definition of Damaged Fuel[2]

"Default" definition of damaged Spent Nuclear Fuel (SNF) - SNF assemblies must be classified as damaged, for both dry storage and/or transportation purposes, if any one of the following conditions exist:

On removal of SNF selected for dry storage or transport from the spent fuel pool, any of the following apply:

1. There is visible deformation of the rods in the SNF assembly. Note: This is not referring to the uniform bowing that occurs in the reactor. This refers to bowing that significantly opens up the lattice spacing.

2. Individual fuel rods are missing from the assembly. Note: The assembly may be reclassified as intact if a dummy rod that displaces a volume equal to, or greater than, the original fuel rod, is placed in the empty rod location.

3. The SNF assembly has missing, displaced, or damaged structural components such that either:

   3.1 Radiological and/or criticality safety is adversely affected (e.g., significantly changed rod pitch).

   3.2 The assembly cannot be handled by normal means (i.e., crane and grapple).

4. Reactor operating records (or other records) indicate that the SNF assembly contains fuel rods with gross breaches.

5. The SNF assembly is no longer in the form of an intact fuel bundle (e.g., consists of, or contains, debris such, as loose fuel pellets or rod segments).

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[2]Derived from ANSI Standard N14.33-2005, "Storage and Transport of Damaged Spent Nuclear Fuel," September 2005.

REFERENCES

1. NRC Spent Fuel Project Office Interim Staff Guidance - 2, “Fuel Retrievability.”

2. ANSI Standard N14.33 2005, “Storage and Transport of Damaged Spent Nuclear Fuel,” September 2005.

3. RE Einziger, CL Brown, GP Hornseth, SR Helton, NL Osgood, and CG Interrante, “Damage in Spent Nuclear Fuel Defined by Properties and Requirements,” Presented at IAEA Technical Workshop on Damaged Fuel, Dec 2005, Vienna, Austria, Proceedings of IAEA International Conference on Management of Spent Fuel from Nuclear Power Reactors, Vienna, Austria, June 2006.

4. MW Hodges, “Status of Technical Issues in Storage and Transportation,” Proceedings of Spent Fuel Management Seminar XXIII, Washington, D.C., Institute for Nuclear Materials Management, January 2006.

5. RA Lorenz, et al., “Fission Product Release from BWR Fuel Under LOCA Conditions,” Oak Ridge National Laboratory, Oak Ridge, TN, NUREG/CR-1773, July 1981.

6. NRC Spent Fuel Project Office Interim Staff Guidance - 22, “Potential Rod Splitting Due to Exposure to an Oxidizing Atmosphere During Short Term Cask Loading Operations of LWR or Other Uranium Oxide Based Fuels,” May 2006.

7. NRC Spent Fuel Project Office Interim Staff Guidance - 11, Rev 3, “Cladding Considerations for Transportation and Storage of Spent Fuel,” November 2003.

Official Transcript of Proceedings

NUCLEAR REGULATORY COMMISSION

Title: San Onofre Nuclear Generating Station Post-shutdown Decommissioning Activities Report

Docket Number: 05000361 and 05000362

Location: Carlsbad, California

Date: Monday, October 27, 2014

Work Order No.: NRC-1224 Pages 1-155

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NUCLEAR REGULATORY COMMISSION

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OFFICE OF NUCLEAR REACTOR REGULATION

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THE SAN ONOFRE NUCLEAR GENERATING STATION (SONGS)

POST-SHUTDOWN DECOMMISSIONING ACTIVITIES REPORT

(PSDAR)

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MONDAY,

OCTOBER 27, 2014

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OMNI LA COSTA

CARLSBAD, CALIFORNIA

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PRESENT:

CHIP CAMERON, Facilitator

DOUG BROADDUS, NRR

LARRY CAMPER, NMSS

AL CSONTOS, NMSS

RAY KELLAR, Region IV

TOM PALMISANO, Southern California Edison

BRUCE WATSON, NMSS

C O N T E N T S

P-R-O-C-E-E-D-I-N-G-S

6:00 p.m.

FACILITATOR CAMERON: Okay, good evening, everyone. My name is Chip Cameron, and I'd like to welcome all of you to the public meeting tonight.

I was going to make a joke and say welcome to the annual meeting of Local 89, but --

(Laughter.)

Okay, but -- all right, all right.

But we do have a serious topic tonight, and it's the decommissioning of the SONGS facility, and specifically, it's a meeting on what's known as the Post-Shutdown Decommissioning Activities Report.

We're going to try not to use many acronyms tonight. We are going to use NRC, for Nuclear Regulatory Commission, and you're probably going to hear the term PSDAR -- that's the Post-Shutdown Activities Report, Decommissioning Activities Report.

That document was submitted by Southern California Edison under the regulations that the NRC has for decommissioning, and you're going to hear a lot about that tonight.

I just want to go through some meeting process items for you so that you know what to expect

tonight, and I'd like to tell you about what the objectives are for the meeting, what the format is going to be, and just some simple ground rules that will help us to have a productive meeting tonight.

In terms of objectives for the meeting, one is to have the NRC provide you with a clear explanation of the NRC regulatory process for decommissioning, and the overview of what's in the PSDAR, the Southern California Edison document.

We also have Tom Palmisano doing a presentation for us tonight to go over the specifics of the post-shutdown report, and we'll be hearing from a panel of speakers including Tom in a few minutes.

So that's one objective, to clearly explain all that to you.

Second objective is to hear any comments, concerns you might have, and to also answer some questions that you might have about the presentations. And our panel reflects all the different organizations that are involved for the NRC in decommissioning, and Larry Camper, our lead senior official, is going to explain that to you in a few minutes.

I am going to have to ask your patience. We are going to have all the presentations before we go out to you for questions so you can get the whole picture,

but we might be here for 45 minutes talking, not at you, but to you, and the speakers are going to try to keep it brief.

But just let me ask your patience with that, and after that's done, we'll go out to you for questions and comments, and I would ask you to hold your questions until all the presentations are finished.

In terms of ground rules, first of all, hold your questions. Second of all, let's only have one person speaking at a time, whoever has the floor at the moment, whoever has the microphone. And that's just very simply so that we can give our complete attention to whomever is talking at the moment.

And I am going to have to ask you to be brief. We're not going to be able to have lengthy presentations by you tonight. We will have listened to your comments, but luckily, there's a relief valve -- if you don't get to say everything that you want to say tonight, the NRC is taking email, comments by email, on these issues, and they're also taking hard copy comments if you want to mail those in, and we'll put that address, the email address and the mail address up for you at -- all during the meeting, so that you can copy that down.

And if your question or concern has already been raised or has been answered, if you could try not

to repeat that -- and I know sometimes that's difficult, because you want to chime in with what you have to say and sometimes it's a little different, but let's try to keep it to what is necessary.

And the last ground rule is just please extend courtesy to everybody, and that goes for all of us, NRC, everybody. Just be courteous. You may hear opinions tonight that are different from your opinions, but just respect the person who is giving that opinion.

And when we do get to the discussion session, after the presentations, I am going to call for people, raise your hand if you have something to say -- I am going to do it section by section, and that is because we are taking camera, web -- and that's going to be for a webcast, and that will be available on the NRC website, so if you want to see what was said, that is going to be your record of the meeting, and also the NRC's record of the meeting tonight.

And I just thank you all for being here tonight, and I'm going to turn it over to Larry Camper.

MR. CAMPER: Thank you, Chip. Welcome, everybody. It's great to see such a great turnout, really. Thank you for taking time out of your busy lives to come and be with us this evening.

We were here last September, and I

commented at the time in my remarks what a beautiful place this is, and I can only repeat it again this evening. You really do live in a lovely spot, so it's good to be with you.

This evening we're going to conduct a meeting that is a required meeting by our regulations, and as Chip said, it's the Post-Shutdown Decommissioning Activities Report Meeting, or PSDAR.

The meeting that we were here for last September was an outreach meeting, as compared to a required meeting. So we're here to gather comments. We do have to share some information with you first, so bear with us as we do that.

Good. I mentioned we were here last September in an outreach meeting. Heard a lot of good input from citizens, and we appreciated hearing all that.

I'll cover the meeting agenda just briefly. In terms of facilitation, you know that Chip Cameron is going to be facilitating. He shared with you the ground rules so we can have a productive and meaningful interface tonight.

It's really about comments. We do have some subjects to cover, but -- we want all of you to be aware of the same information at the same time, but it

really is about getting comments from all of you here, or as many of you as want to speak.

We do have some meeting feedback forms outside on the table. I would ask you to take time to complete the meeting feedback form. We like to get some feedback as to how the meetings went, and did you find it valuable and useful.

We will be adjourning at 9:00 p.m., we have the room contractually until 9:00 p.m., so we do have to stop at that point in time. If we go over a few minutes, I don't think the hotel will come in here and run us out, but let's strive for that 9:00 p.m. objective.

During the September meeting, it was a meeting that we initiated although it was not required by our regulations. And we did that because there was a lot of things that had happened at this particular plant, and our agency, our chairman in particular, our senior management, all had an interest in seeing to it that we came out here, heard your concerns, and communicated about not only the decommissioning process, but anything that folks wanted to talk about, and we did that.

At that time, we had a government-to-government meeting with local elected

officials during the day, we did that again today. And we also had a meeting with non-government organizations at that time, and today we met with representatives from the Community Engagement Panel.

Each of those meetings about an hour and a half in duration, and we found them to be indeed very useful, with much good input. And I would -- I think it's fair to say that the elected officials and the Community Engagement Panel is listening. They shared a lot of very good information with us.

Last September, there were a myriad of topics that were talked about. I put down two in this particular slide, fuel management -- and I am sure we'll hear about fuel management again this evening -- as well as decommissioning timing, when will decommissioning start, how long will it take, what are the guidelines and the regulations by which they have to carry out their decommissioning activities?

For tonight, I want to start out by saying that I'm going to mention each of the speakers that are up here at the table tonight. Each of these will be making a presentation.

Chip alluded to the fact that our responsibility for decommissioning is spread out in several places, so we wanted to make sure we had the

right expertise here. But I also want to point out -- and I'd ask the NRC staff here, if you'd stand up for a moment and turn around and let the folks see you -- we have a number of NRC people here, and the reason for that is because we wanted to make sure we had the expertise here. Thank you very much.

And if some question comes up over the course of the evening that one of the staff feel that they can better answer, then they can certainly indicate, and we'll recognize them, and they'll do that.

Also, we'll take a quick break, and we'll be here just for a little bit after the meeting. Any of these folks are a source of information for you, especially if you're shy about getting up and asking questions or making comments during the program this evening.

First is Bruce Watson. Bruce will address the Post-Shutdown Decommissioning Activities Requirements.

Next to him is Doug Broaddus, and Doug will cover the review, our review of the PSDAR, and the licensing status of the facility at this point in time.

Al Csontos will address spent fuel safety.

Ray Kellar, who is with Region IV, will address our inspection program.

And last but not least is Tom Palmisano, who is actually with the SONGS unit and will provide the contents of the PSDAR itself.

I did also want to mention that of our staff sitting out here, Keith McConnell, Dr. McConnell, was the manager that was in charge of our recent waste confidence decision, our long-term storage rulemaking, and so Keith is prepared to answer some questions about that initiative if need be.

Chip will lead us into our public comment session, and then at the end I like to make some summary comments and share what I call aha moments, things that we heard that were a big deal.

So let me say that it's about our mission. Our mission, as you see on the slide, is to regulate the nation's civilian use of radioactive materials to protect public health and safety, promote the common defense, security, and protect the environment.

That's our mission during operations. That remains our mission during decommissioning. The entire decommissioning process will be carried out consistent with our regulations and our continuing regulatory oversight to achieve the objective that you see there on the slide defining our mission.

I would also point out that there is a great

deal of interest, of course, in things environmental, environmental impacts. There is a law called the National Environmental Policy Act of 1969, and we have regulations in our regulatory Part you see there, Part 51. This is where we carry out our regulatory responsibilities to fulfill the requirements of the National Environmental Policy Act.

An important part of that is that the PSDAR contains an updated environmental report, and Tom will speak to the contents of that during his remarks.

And then of course ultimately, when we ultimately receive a license termination plan for this facility, we will be conducting an environmental assessment as part of that licensing action.

Our regulations for decommissioning are a set Part in Part 20, so Part E. What's important is to -- what does the term mean. The term means to remove as a facility safely from service, and reduce radioactivity to a level that permits release of the property for unrestricted use and termination of the license, or release of the property under restricted conditions and termination of the license.

I would point out that no nuclear power plant to date has ever pursued the restricted release pathway, and the facility for SONGS 2 and 3 is not doing

that either, they're pursuing unrestricted release.

In terms of the release criteria for unrestricted release, in order for a site to be suitable for release in an unrestricted manner, the dose that must be achieve is a total effective dose equivalent, or TEDE, of equal to or less than 25 millirem, and as low as reasonably achievable.

And achieving that, that means the dose to the average member of the critical group, all pathways, including groundwater, and the period of performance is for 1,000 years.

You also see below it the dose criteria for restricted release. I would note that the criteria is the same except that there are provisions in place for institutional controls, and some criteria if those institutional controls fail, but that's not being pursued here.

Now when you see or you hear 25 millirem, what does that mean? Well, let me give you something to think about.

Millirem is a dose unit of exposure to people, radiation equivalent in man is what millirem stands for.

If you get on an airplane in Los Angeles and you fly to New York, you get about three millirem. The

natural background radiation in the United States, depending upon where you are, ranges from about 300 to 600 millirem.

There are places in the world, though, where the natural background radiation is much higher than that. Saskatchewan, Canada comes to mind, about 4,000 millirem per year.

So at least when you hear that term now, 25 millirem, you'll have some idea of what we're talking about.

I would point out that while our criteria is 25 millirem and as low as reasonably achievable, the decommissioning of nuclear power plants to date have all achieved a level of exposure much lower than 25 millirem, on the order of a few millirem, four, five, six millirem.

This is a very important slide for you as members of the public. In fact, I think it's probably the slide that, if I were you, I would be most interested in, because it tells you what happens in the process, and where in the process you have informational awareness, or where in the process you get a chance to make comments.

On the left you see what the licensee is required to do. In the center is what our agency does.

And on the right, of course, is the public.

There's an initial notification that takes place. It's a cessation of operations, and also a certification that the fuel has been removed from the reactor vessel.

Then, there comes a decommissioning report. That is the PSDAR. That is why we are here tonight.

Now you note that we review that, and we conduct a public meeting near the site, and you, the public, have an opportunity to comment.

I do want to be clear, though, that we don't approve the PSDAR. We review the PSDAR, and the licensee has to wait 90 days while we do that, while we carry out our review to ensure that our regulations are being satisfied, that the PSDAR is in fact adequate to satisfy those regulations. We also go about the comment-gathering process, and we will consider your comments as we continue to carry out our review.

Then actual decommissioning takes place. The reactor is completely decommissioned over a period of time, and Tom will specify tonight in his remarks how long this utility plans to take to accomplish that objective.

And then the next major milestone in the

process is a submittal of a license termination plan. The utility will submit that license termination plan approximately two years before it desires to terminate its Part 50 license, that's the Part of our regulations that we use to regulate power plants, or to shrink the footprint to the remaining independent spent fuel storage installation pad.

That process, that LTP, is something we do review, and either approve or deny, and I think very importantly for your awareness is that the LTP process is a licensing action that carries with it the opportunity to request a hearing.

And if parties, or a party, requests a hearing and achieves standing, then we will carry out a hearing through our adjudicatory process.

Then final decommissioning takes place, a number of final surveys are done, we conduct verification surveys, and all that information is available to you the public through the inspection reports that are generated as we go about monitoring the Final Status Surveys.

And the Final Status Surveys are designed to achieve that dose standard that I cited a moment ago.

This particular slide, the point of showing you this slide is to show you that we've decommissioned

a lot of sites. We've decommissioned on the order of 80 sites, including 11 nuclear power plants.

We have a lot of experience, we have a lot of expertise, and not to sound puffy, because that's not my intention, but I just want you to know we have a lot of experience. We've done this a number of times. And we're going to bring all that expertise and experience to bear as we go about monitoring the decommissioning of this facility as well.

There is a transition that goes on when we move from an operating power plant to a power plant in decommissioning. We have a number of program responsibilities. Just so you know who's on first, I put this slide in. It shows you that the Office of Nuclear Reactor Regulation continues project management until the post-shutdown defueled technical specifications are issued, Doug will mention that a bit in his commentary.

Then the project management is transferred to my division within the Office of Nuclear Material Safety and Safeguards, and we will shepherd it through the entire decommissioning process until the very end.

Inspections continue to take place, and that's transferred to the Division of Nuclear Materials Safety and Safeguards from the Division of Reactor

Projects, and of course, Region IV will carry out that inspection activity, and you'll hear about that from our Region IV representative.

And then, of course, support continues from the Nuclear Security and Incident Response Program within NRC.

So I will stop there. Bruce Watson will follow me and talk to you more about the PSDAR process. Bruce?

MR. WATSON: Thank you, and good evening, and thank you for all sharing your evening with us.

The NRC regulations which Larry went over that carry decommissioning have been in place for over 17 years. Larry went into a little more detail on the Part 20, but the key factor for the reactor decommissioning is in Part 50, specifically 50.82.

These regulations took into a lot of experience that we incurred over the years with some of the initial decommissioning of some reactors in the early 1980s and 1990s.

Some of the key milestones from SONGS so far have been that they ceased operations -- permanently ceased operations on June 7, 2013. We received the defueled certifications for the two units also during the summer of 2013, and on September 23, they submitted

the PSDAR for our review.

We issued a public notice for that PSDAR, and of course the public -- the PSDAR is available in ADAMS, which is our agency-wide document system, and there's the number for putting that in off the public website.

Our guidance allows three types of decommissioning options. The first is DECON, which basically means you're going to do prompt remediation, you're going to begin fairly quickly after you transition the reactor to a state where it can be decommissioned.

That transitioning typically takes one to two years, draining systems, isolating systems, isolating electrical systems and making the plant safer for disassembly.

The second option is SAFSTOR, in which you can allow the plant to stay in a sort of a mothballed state for a number of years.

And then ENTOMB, which we currently do not use and do not have any regulations for and no one has requested, and we don't expect anybody to request entombment, is another option, but not currently available.

But the bottom line is radiological

decommissioning, and I stress the radiological decommissioning, which is what we regulate, the radioactive part of the decommissioning has to be completed within 60 years.

Now San Onofre has chosen to do DECON in their PSDAR, so they're going to begin fairly quickly with their schedule that's in there.

With the PSDAR from SONGS, though, they meet the three criteria that we're reviewing. The first part, major requirement, is a description and schedule for the planned decommissioning activities, so it as a fairly high-level schedule on how they plan to accomplish the decommissioning.

The purpose in this is to allow the NRC to schedule our resource to support inspection activities throughout the decommissioning to ensure that we're there for major activities where we think we should be there to ensure that the work is done safely.

The second item that's required to be in a PSDAR is a site-specific decommissioning cost estimate, including the costs of managing the irradiated fuel. In most cases, this is a fairly extensive document and has good detail in it.

The third thing is that the PSDAR must require -- is required to provide a discussion that

provides the means for concluding that the environmental impacts associated with the decommissioning will be bounded by the appropriately -issued Environmental Impact Statement.

Now, we have a document called NUREG-0586, it's publically available. It is the generic Environmental Impact Statement document where you can look at all the things that have been previously reviewed.

The PSDAR regulations require that we hold a public meeting, which we are doing tonight. We are here to hear your comments and hopefully answer your questions. Like I said, the PSDAR is available for public comment. You can find it, again, at that ADAMS number. We also passed out a number of CDs for your use.

We will accept written comments, like Larry said, and also ones from the website.

The key thing here is that the licensee can begin major decommissioning 90 days after the PSDAR has been submitted to us.

Lastly, I wanted to mention that we have continued our transition activities. One of those transitioning activities is in the inspection area, and Senior Inspector Greg Warnick has been here a little over a year now, and will be here for the near term, as

part of that -- ensuring that the plant continues to be transitioned safely.

And with that, I'd like to turn it over to Doug Broaddus, who is going to talk about the reviews for the PSDAR.

MR. BROADDUS: Thank you, Bruce.

All right. So I am here tonight to talk to you tonight about our review process for the PSDAR, but I am also going to talk about some of the other review activities that we have ongoing associated with SONGS, both their licensing actions that they have, and some other related activities that we have under review.

As Bruce indicated, the -- our primary review is, you know, does the PSDAR contain the information that's required by the regulations? And Bruce talked about specifically those requirements.

The other area where we use as review guidance is in our Reg Guide 1.185. It's a Guide that provides the standard format and content of the PSDAR. So we -- that provides more details as to what information we expect to have in the PSDAR.

So the -- our review process, then, is that once we receive it, our project manager for the SONGS facility will send that out to all, and coordinate with the various different technical expertises that are

needed to look at the various different issues within the PSDAR.

And our general process is, if there's additional information that we need to either confirm that it meets the requirements or that there's specific information that's missing that we need, we'll ask for additional information of the licensee.

So what is our critera? What do we look for specifically from the standpoint of the PSDAR?

There are a number of things that could cause us to either need additional information on the PSDAR, or in fact, to possibly find it deficient.

One is that it doesn't contain the information that's required in the regulations, if it doesn't provide all the information that's specifically required.

The other is if the costs that the licensee have indicated would exceed the costs -- or the funding that they have available to them.

Another would be if they are -- if they haven't fully described the process that they're going to follow, or the process that they have described is one that could not actually be implemented. So we would -- that could cause us to have problems with that.

The other would be that the schedule is --

would not be completed within the 60 year required time frame that Larry mentioned earlier.

The one caveat to that is if the NRC determines that there's a specific reason why a longer than a 60 year period would be necessary, for example to protect public health and safety for a particular reason, then that time period could be extended, but that's something that we would have to specifically review and authorize the licensee to go beyond the 60 years.

All right. And the last thing, obviously, and the most important, is there -- are the activities going to endanger public health and safety?

One example would be if there's no waste disposal facility available, do they have to -- would they -- well actually, I'm sorry that's for the 60 years. If there wasn't a waste disposal available for disposal of the material, it could be that they need to go beyond the 60 years.

But from a health and safety standpoint, that's part of our criteria, just to make sure that the activities that they're doing are in compliance with our regulations and aren't in violation of our health and safety requirements.

All right. Next is, as part of the PSDAR,

a site-specific decommissioning cost estimate has to be provided, and this decommissioning cost estimate has to address the specific -- the overall cost of the decommissioning. It has to provide a good understanding of the costs, the basis for those costs, and it has to ensure, you know, provide enough information to us to be able to say that they have reasonable assurance that they have funds available to complete their decommissioning activities.

The other thing is if the licensee has to change their plans, or if they have unforeseen problems as they go through the decommissioning process, that they have a means of adjusting their cost estimate for identifying new, or identifying new funding mechanisms if they don't have enough money to pay for it.

And that's -- from that perspective, we've heard the question of, so what happens if they get into that type of situation?

So the licensee is required to maintain the level of funding. If they have to change their plans, they have to revise their cost estimate in the PSDAR and submit that to us for a review, and we would look at that to ensure that they've come up with another funding mechanism or some other type of mechanism, or they've changed their approach to ensure that they're going to

have the funding available to complete the decommissioning activities.

And these -- the licensee also has to submit a revision every year to this decommissioning cost estimate, so we're going to see changes as they're happening through the years, and so any problems that would occur, we would see them well before they would ever happen, and we would ensure that the licensee is taking action to adjust those.

As Larry indicated before, NEPA is part of the environmental process, is part of our review. In this case, the -- what the licensee would address in their PSDAR is that they've looked at the environmental impacts of the activities that they're going to take, they've looked at the previous environmental assessments that have been conducted, both for the site -- a site specific assessment that was done for the site as well as the generic final environmental assessment for decommissioning facilities -- looking at those to determine whether or not the environmental impacts they are going to incur for their activities would be bound by those prior estimates.

And so they would have to provide the basis for why they believe that it's met within, that it's within those prior environmental impacts.

All right. And the inspection we do have, as Bruce mentioned, we have an inspection program where they'll go out and they'll confirm that the activities they're doing are in accordance with the prior environmental impacts and that they have the appropriate analyses to support the fact that they don't believe that their activities are going to exceed those prior environmental impacts.

All right. So we are here tonight to get comments back from the public. We are also going to be, as Chip indicated, we are going to be receiving additional comments for the 90 day period after we have gotten the PSDAR, which is through December 22.

We'll take those comments and we'll look at them, we'll consider them in our review as we're reviewing the PSDAR.

In particular, we would be looking to determine whether or not there is any health and safety impacts that would be encountered by their plan.

If we have questions, we will ask additional information from the licensee, we will ask them to provide that additional information.

We -- as Larry indicated, we don't approve the PSDAR, but we will document the completion of our review and determining, if we've determined that

there's no additional information necessary, and we'll document the resolution of the comments that we would receive.

One thing that's important is the licensee cannot start major decommissioning activities until at least 90 days after the PSDAR has been received by the NRC. That gives us the time period to review the document, to receive public comments, and to be able to review those and act on them.

All right. One of the other areas that we are reviewing, it's called the Irradiated Fuel Management Plan.

Within two years after a licensee shuts down, they are required to submit to us an Irradiated Fuel Management Plan that addresses the costs of managing all spent fuel that's at the facility.

And this is not -- it's not -- it's a separate document from the PSDAR, but there is a relationship between the two, because the Decommissioning Cost Estimate has to include the cost of managing the spent fuel.

So although we're not asking specifically for comments on the Irradiated Fuel Management Plan, I wanted to make sure everybody understood that we are reviewing that as well. And that document does need --

was -- is required to be submitted for our review and preliminary approval.

Our review on that will also look to ensure that they've got a good plan for managing the irradiated fuel and that they are able to account for the cost for that as well, that's part of it.

And the other part of it is to ensure that their plan complies with all the requirements for possessing and storing irradiated fuel.

I think that's it. I'm sorry, I almost missed one.

So there are other licensing actions that we're also doing for the plant right now. So as the plant, once it shuts down, the plant transitions from an operating status to a decommissioning status. So to do that, the licensee comes in and requests a number of licensing actions.

Those licensing actions can either be submitted as a license amendment, they could be an exemption from a specific part of the regulations, or they could be a request to rescind an order or have relaxation of an order.

So the types of requests that we've received are for changing the staffing levels of the plant, to change the type of personnel that they'll have

there to focus more on the spent fuel and decommissioning activities; changes to the license condition to address, to more focus on the managing of the fuel activities; there's also been requests to change the emergency preparedness activities given the shutdown condition and the defueled condition of the plant; and we've received requests for previous orders for the licensee to, also to rescind some of those previous orders.

So those are the types of licensing actions that we've had, where we've approved a couple of them -- we're still in the process of reviewing a number of other ones, and we understand that the licensee is probably going to submit some more to us, and so I can't speak to those yet. So that's it.

MR. KELLAR: Good evening, I'm Ray Kellar, I'm from Region IV.

You've heard a little bit from Bruce and Doug, and you will be hearing from Al, about the regulations, the reviews for the licensing conditions, and also about safety reviews and evaluations here in a little bit, but I am going to talk about how it all comes together a little bit, and I am going to talk about the inspection program, particularly in Region IV, from a decommissioning and a spent fuel safety standpoint.

So how do we get safely from a operating plant to a plant that's been fully decommissioned? As Larry mentioned, there's been the opportunity for the Nuclear Regulatory Commission to have oversight of, I believe he said, 11 nuclear plants over the last recent period.

I know in Region IV, we've had several opportunities for decommissioning plants, specifically Trojan, which is near Portland, Oregon; SONGS Unit 1; and we're currently inspecting Humboldt Bay, which is in Northern California.

So how do we assure compliance? The NRC looks at the regulations, which is what we've been talking about; we look at the licensing-based conditions, including the technical specifications, decommissioning technical specifications; also guidance documents, which would be things such as NUREGs. These are all part of the licensing basis which are a part of the inspection when the inspectors go out to do their work.

Also, the safety reviews will be performed. Al will be talking some about the reviews, I am sure, relative to spent fuel storage. And that leads down to the NRC inspection activities, where the inspectors go out using the regulations, using the licensing-based

documents, and they perform the inspections to ensure that the licensee is doing things in accordance with what is safe and what they have been approved to do.

If we find that something is not right, then we will look at the NRC enforcement program and evaluate whether or not that would come into play.

So the regional-based inspection activities will consist of inspection of the spent fuel pool. We will generally look at the higher-risk activities, such as moving the fuel, such as loading the fuel into casks. Those will get extra attention.

During and after remediation activities, we will make sure that we're doing independent measurements to confirm the licensee's survey methodologies.

Additionally, inspection activities will include the spent fuel, that's also in my branch, as well as a movement to the (inaudible), which includes a series of operational tasks and oversight of those activities.

Physical security, while it's not my branch, will still be inspected according to the requirements. And there are a number of inspection plans that include oversight of activities including licensee management and organization; review of the

license safety evaluations, so making sure that their 50.59s have been done correctly; reviews of licensee self-assessments and audits; radiation exposures during decommissioning, transportation, and rad waste, making sure that all the requirements are met for that; as well as effluent environmental monitoring.

So the objectives of the inspection program include the independent verification of the safety of the licensee's activities, the adequacy of the licensee's controls and programs, ensuring that safety programs are promptly identified and the licensee puts effective corrective actions into place, identifying violations that occur and utilizing the agency-enforced policy to make sure that those are disciplined correctly, as well as review and examine any trends that develop dealing with the licensee's safety performance.

Using the inspection procedures, we planned some of the activities up to a year in advance, and the inspectors know what portion of those will occur at what time.

Those activities are coordinated with headquarters. We sometimes bring technical expertise along with us to the site to assist in those program inspections. They may be announced or unannounced, quite often they're announced, but we have the

opportunity to do an unannounced inspection if there's a need to.

We utilize inspection plans that document those procedures and those activities that we want to perform inspection activities of. We have exit meetings at the end of each of the inspections where we do an exit with the licensee and present their findings at that time, and if there's any potential enforcement actions, those are also discussed.

The inspection reports are normally issued 30 days after the exit, that would be for a -- not a team, but a team inspection consisting of three or more inspectors would normally be 45 days after the exit with the licensee before that report is issued.

The NRC enforcement policy, there is a link at the bottom of this slide that shows how you can get to the enforcement policy if you're interested in how the agency performs the enforcement activities.

So what happens after the inspection? After the inspection, the inspectors come back to the region. They brief the branch chief, and there is a debrief that includes NRC management where the findings are presented. And if enforcement actions need to be taken, those are also discussed, along with the enforcement officer, and those activities are followed

through with.

The inspection reports are available in ADAMS, those that are available publically. Security reports, the details are not available, but the cover letter is usually available with ADAMS.

You can go to the link that I show on the bottom half of this web -- this slide, and it shows how you can get to the ADAMS website, the document numbers -- docket numbers, are shown here as well, you can do a search using Advanced Search under those docket numbers using the website for ADAMS, and you can actually look up the inspection reports that way. And we also track and follow-up on inspection reports.

And that concludes my portion.

MR. CSONTOS: Hi, my name is Al Csontos. I am the Chief of the Renewals and Materials Branch in the Division of Spent Fuel Management in the Office of Nuclear Material Safety and Safeguards.

What does that all mean? That's a long-winded title. What we do is we develop and implement the licensing certification and inspection of spent fuel storage facilities, the dry cask storage systems, the radioactive material transportation packages, for the -- with the applicable regulations, 10 CFR Part 71 for transportation, 10 CFR Part 72 for

storage.

We also coordinate with federal agencies like the Department of Transportation, for example, for some of the transportation packages' certifications. Four international bodies, regulatory bodies, as well, IAEA, International Atomic Energy Agency, and our other colleagues around the world.

Some of the things that we've learned in what we've done has been from our interactions with these international participants, and also Native American tribes.

So what are these things? These are spent fuel storage casks. Over here at SONGS, you have the lower-left version. It's a NUHOMS System 1029, okay, certificate of compliance.

What they are is that once the fuel has been cooled in a spent fuel pool that are part of the reactor, it is loaded into these special canisters, for these canisters, they're stainless steel canisters, they're about 5/8ths of an inch thick, and they're designed for the specific fuel that's at these sites, these reactors -- either a pressurized water reactor or a boiling water reactor.

The canisters are then vacuumed down, take all the water out, and they are backfilled with inert

gas. That inert gas is usually helium, and it has a purpose of helping with the cooling of these canisters, and also to -- just to stop any kind of corrosion from occurring on the inside of the canisters.

So they're welded, they're vacuum-checked, they're leak-checked -- when they're vacuumed down, they're backfilled, and then they're checked for leaks, helium leaks. And then once they've passed all those rigorous tests and all the fabrication requirements and the quality assurance procedures, they are put into these casks.

And what these are, these concrete -- does this have a -- so right here, that's about five feet of concrete. And so that helps with the shielding, the bioshielding from radiation, as well as structural, for seismic concerns.

And so these are the ones that you have at SONGS presently. There is a third one, I know that you all were working with two vendors recently, and there's a third one that's -- this is a vertical system that multiple companies develop, and there's a third one that's an underground system that I didn't show here.

So we have a total of 71 total sites in the United States by this map. These are independent spent fuel storage installations.

There are, in 34 states, there are over 2,000 casks that have been fabricated and are in place, with over 82,000 fuel assemblies. They are all across the country. There's 56 in -- this is, it gets into some regulatory jargon here, but 56 General Licensed Sites and 15 Specific Licensed Sites.

And there are seven at decommissioned sites that will be similar to what is here at SONGS.

What is the difference between a general license versus a site-specific license? A general license is only granted to Part 50 licensees, reactor licensees. They require us at NRC, in our division, to review those cask designs.

We review them for certain things that I'll go into in a little bit, and then we do the site evaluations and such, and then the site-specific cases are specific to Part 50 reactor licensees or other applicants that would like to have -- would like to store this waste at their site.

So, like, for example at Private Fuel Storage in Utah, that is an SSC, a site-specifc, SSC, has no reactor there, but they were willing to take fuel if they were allowed.

So Part 72 regulations. What are our review areas? What do we do in our daily jobs? We

review the general design criteria, making sure that they meet the requirements for offsite radiation dose, subcriticality, and confinement.

What does all that mean? Confinement, we retain and make sure that the radionuclides in the fuel stay within the canister. For subcriticality, we don't allow any type of -- we make sure that the fuel is sitting in there so that they never get a place where they can go into a critical state. And offsite radiation dose, we have sensors around the associated pad to make sure that there is no dose that is beyond the requirements, the regulations.

We also have quality assurance programs, that's what Ray was talking about earlier. We fabricate these canisters -- or not we, but the licensee buys them from a fabricator -- and we make sure that they're done in a quality fashion.

Physical protection security, sighting, reporting requirements, and then training and certification of the personnel at the sites. We review all of those.

This takes a lot of people. And what we have is we have a cadre of subject-matter experts. We have structural engineers, we have thermal engineers, we have criticality specialists, we have health physics

folks who do the shielding and radiation protection.

We have materials folks like myself who look at all the materials issues, the corrosion issues, all the material properties issues, making sure that the confinement is maintained, and also making sure that the quality assurance folks are out there. Those are the inspectors, in a lot of ways.

So, what do we also look at? We look at three different types of scenarios -- normal, off-normal, and accident conditions. Normal and off-normal, normal occurs regularly, repeatedly. Off-normal is occasionally, once a year, maybe. And then accident conditions are things that we make sure that the licensees, make sure that these are fabricated to withstand these types of events -- tornado winds, missiles, earthquakes, seismic activity, floods, tsunamis, and fires and explosions.

Again, the technical reviews are quite long, and a lot of our reviews for new applications can be two, three years, okay?

We have structural engineers who look at the confinement that are maintained under all these accident -- the accident, normal, and off-normal conditions. We have criticality folks who look at making sure the fuel is sub-critical in all these three

cases. Shielding that meets the radiation dose requirements at the site boundaries -- and then the thermal, making sure the cladding temperature is maintained so we have a defense in depth, so we can rely on the cladding.

And then the materials folks, who look at the property or properties, and also aging effects, during the renewed operating period, and I'll go into that a little bit later.

So for transportation of these systems, the same technical reviews as for storage. There is a little bit more in terms of, you need to show us, to the licensees that are out there, you need to do a bunch of tests.

Those tests are drops, 30 foot drops, they're puncture tests, they're deep water immersion tests, making sure that nothing will ever come out of these canisters. Fire -- there is a 30 minute requirement for jet fuel, making sure it doesn't fail in that way -- and all these other normal types of transportation, vibration loads, small drops, heat and cold.

Cold is a big deal. Some licensees have been rejected because they can't meet the requirements for transportation at -40 degrees Celsius.

So now, storage renewal. So what are we doing now, the things that are changing on the landscape here.

There are two periods. There is the initial licensing period, which is the first 20 years. And then after that, it's between 20 and 60 years is the renewed operating period, okay?

The renewal, renewed operating period, has been going through some updates of late, and so what we've done is we've established a group at the agency, a strategy team, and what we've done is we've been trying to look at what we can do to update the guidance, because what we're looking to do is we're looking to change our approach to spent fuel management in the renewed period of operation to an operations-focused aging management, one that is learning from our operational experience.

It's proactive to get ahead of issues that, possibly, for degradation aging effects of canisters. You all know that your cars don't last 100 years, or 50 years. And they age. And the same thing happens to canisters, or anything that's manmade, and so we're trying to get ahead of these, and we're trying to be proactive and have a responsive aging management, so if we do find anything, we go over and fix it, or require the licensee to fix it.

So what that requires is aging management programs that we're developing in-house, and we're developing these guidance documents out there to help the licensees with how to manage aging as time goes on.

So two issues that have come up that have been recently dominating a lot of our discussions.

One is cladding integrity and high burnup fuel, in particular. And the second is chloride-induced stress corrosion cracking, which I'm sure we'll hear a lot about today.

We have spent at the agency over $9 million over the past eight, nine years. We have spent an inordinate amount of staff resources on these issues. We feel we're getting ahead of these issues now, okay?

We're developing aging management programs for both, and we're -- we'll discuss in a little bit, and you can ask a lot of questions, but in 1927, NUREG 1927, that will be coming out this springtime sometime, and we will have these aging management programs, these draft ones now, but they will be placed in there, to show how we are going to manage these aging effects.

But ultimately, we believe high burnup fuel is safe for storage and for transportation. We have licensed both high burnup fuel for storage and for transportation. Recently, we had the MP-197 that we've

approved for transportation casks for use of high burnup fuel, okay.

We have other research that we're doing right now, at Oak Ridge National Labs, and we also are also working with the Department of Energy for their Cask Demonstration Surveillance Project.

We have an information notice on chloride SCC, chloride stress corrosion cracking, I hate to use acronyms.

And we're also developing this aging management program that's part of the NUREG 1927 that we can talk about at length if you wish.

So in summary, our regulations in 10 CFR 71 and 72 ensure safety for both storing and transporting spent nuclear fuel. We have a large group of people who have a multi-disciplinary technical review for all of the different issues that I identified, and we are maintaining confinement under routine and accident scenarios for these canisters.

We're going forward with an operations-based aging management program, and it's going to be a learning, proactive, and responsive program for renewals and for aging management.

And you can read the rest. We're creating a stable, predictable, and efficient renewal regulatory

framework, with clear, open, and transparent, reliable, regulatory expectations to the licensees to manage aging.

I think that's it.

MR. PALMISANO: Good evening. I'm Tom Palmisano, Vice-President and Chief Nuclear Officer for the San Onofre Nuclear Plant.

I've seen a number of you before in community engagement panel meetings, and I am going to cover the specific Post-Shutdown Decommissioning Activities Report that we submitted, and summarize what's in the report, talk about the different periods for the decommissioning plan.

I'll summarize the Irradiated Fuel Management Plan, and I'll also summarize the Decommissioning Cost Estimate.

So before I get into the specific remarks, Southern California Edison and our other co-owners issued some decommissioning principles early this year to provide clarity and guidance in terms of the principles we will follow in developing the plan and executing the plan to decommission San Onofre.

Particularly, our principles are safety, stewardship, and engagement -- safety first, safety in terms of all the activities, safe storage of spent fuel;

stewardship in terms of being good stewards of the environment, good stewards of the decommissioning trust fund, recognizing this fund has been contributed by our rate payers; and engagement and transparency.

The Community Engagement Panel, which a number of you have been at the meetings, is part of our effort, our outreach effort, to engage the community, to be transparent in what we intend to do, and to listen actively to feedback.

I would encourage you to visit songscommunity.com. All the documents I am going to discuss tonight have been posted throughout the course of the last six months on songscommunity.com, so they're not only available on the NRC website, but they're readily available on our website, along with more detail.

So just to recap, you've seen some of these time frames before in the NRC presentations. June of 2013, we decided to permanently shut down both units 2 and 3. We certified permanent cessation of operations, one of the regulatory requirements.

We also at that point certified we had permanently defueled. The dates, I didn't put those dates on this slide, they were on the NRC's slide.

Through the course of 2013, we transitioned

from operating plant staffing to decommissioning staffing, and in January 2014, we developed a 20 year decommissioning plan that we submitted as part of the Post-Shutdown Decommissioning Activities Report.

As you've heard, by NRC regulations, we have up to 60 years to complete the radiological decommissioning of the plant.

As we look at the situation at San Onofre, as we listen to feedback from our stakeholders, as we look at where our trust fund is, we think the right thing to do is to go into the prompt decontamination and dismantlement process and decommission the plant in basically a 20 year time period, and I'll show you a little more detail in a minute.

And then in September 2014, we submitted the required decommissioning documents, the Post-Shutdown Decommissioning Activities Report, the Irradiated Fuel Management Plan, and the Decommissioning Cost Estimate.

So let me give you the highlights here. Those of you who have been at the Community Engagement Panel meeting have seen this slide before. This is a simple look at the 20 year timeline for decommissioning the plant.

The top line is not to scale. To the left

of the bold vertical line is between now and January 2016, so we are not in the major decommissioning phase at this point. We need to submit the NRC documents we have done. So everything we're doing to the left of that line is planning and preparatory activities, and the licensing submittals to change our design and licensing basis.

There's a variety of physical plant changes we make such as defueling the plant, draining systems, eliminating non-radioactive hazards.

There's the licensing submittals, this was referred to as some of the other licensing actions we've requested, for example defuel technical specifications that changes the conditions of the license to match the fact that the plants are defueled; the specific decommissioning submittals that I've talked about; and then the dry fuel storage system.

I am going to talk about dry fuel storage more in a minute. We already have a system on site with 50 canisters loaded with fuel that needs to be expanded to allow for the decommissioning process.

So we're in the process, and we've discussed this extensively at the Community Engagement Panel, of selecting the best plan to expand the system and load the remaining canisters.

Then to the right of the heavy bold line is actually the balance, first of all, of the decontamination and dismantle period. It's about a ten year period where we do the major decontamination and dismantlement, and then farther to the right, you see words like completion of remaining site restoration work -- that's the non-radiological site restoration, as well as the License Termination Plan.

On the NRC slides you saw, that's the formal process where we submit the License Termination Plan, submit the Final Survey Plan, and execute that plan subject to NRC review and approval.

And then at the end of that 20 year period, the plant is decommissioned, it's dismantled. What will remain is an independent spent fuel storage installation, and that will remain until the Department of Energy starts taking fuel off the site, and we'll see more about that in a minute.

Post-Shutdown Decommissioning Activities Report, we've heard a lot about that, so I'll just summarize this very quickly.

We have submitted a report that meets the regulatory requirements. It describes the planned decommissioning activities. It provides the schedule for the activities. It summarizes the expected cost.

And it has the discussion of the environmental impacts and concludes that our planned decommissioning activities are bounded by the site-specific environmental assessments that exist and the NRC's generic Environmental Impact Statement for decommissioning.

So our Post-Shutdown Decommissioning Activities Report meets the regulatory requirements. It is under review by the NRC.

So I want to go in a little more depth in terms of what the three elements of the PSDAR are. The license termination, and that is equivalent to radiological decommissioning; the used fuel management program; and site restoration, think about that as the non-radiological decommissioning to finish restoring the site.

This next graphic is just a bit hard to see. This actually runs out to 2052. I am going to show you more detail that's more readable here in a minute.

The green is going to be the license termination periods, or the radiological decommissioning periods; the orange in the middle, or the yellow in the middle, is actually the used fuel management periods that will go through the first 20 years and then extend out as far as 2052; and then the

red is the site restoration period, think about this as the non-radiological decommissioning, once the NRC aspects of decommissioning are complete.

So we've created some graphics based on some feedback from the public and the Community Engagement Panel to try to make this a little more visible.

Again, all these slides will be publically available on our websites, and I'm sure the NRC will post these as well.

So, and I've also tried to indicate the time periods and the types of activities.

So we're talking about license termination or radiological decommissioning on this slide. If you look from the left, June 2013-December 2013 defueling both reactors, certifying the reactors were defueled.

We're actually in Period 2 now, where we're doing the initial planning. And the initial planning is essentially complete, it's relatively high-level planning, and we're making the NRC submittals.

So this is the initial planning phase where we identify what the 20 year decommissioning plan is, the expected cost, and the summary of the activities.

Once we enter the major decommissioning phase, which we are not in, starting as early as July

of 2015, we'll start major decommissioning activities.

This will be system decontamination to reduce the radioactivity's level inside systems. This will be starting the planning, and then as we get into later in the period, removing the internals of the reactor vessel. This, other than the fuel, is the most radioactive part of the plant, done underwater. This is done early in the process.

This would be something 2017, late 2016, 2017, would be a rough timeline for that to start.

And later in this period, and I'll talk about spent fuel more in a minute, we will be offloading the spent fuel pools, actually moving fuel likely in the 2017 time frame to an expanded ISFSI (phonetic) system.

Period 4, then, once the fuel pools are offloaded, we'll get into the system-enlarged component removal. At this point, we've reduced the level of radioactivity in the plant, we've moved the fuel out of the spent fool pools into the ISFSI or dry fuel storage system, and then we'll start removing the major components inside those reactor containment buildings and the other safety-related buildings.

At the end of that period, then, October 2022 - July 2024, we're in a period where we're decontaminating the buildings, once the internals have

been removed, and ultimately in Period 6, 2024 to 2032, we're actually demolishing the buildings themselves.

So that's roughly what a 20 year plan from 2013 to 2032 looks like, for the radiological decommissioning of San Onofre.

Let me move on to used fuel or spent fuel storage. So this is a companion timeline. Again, we're in the planning period. We're now in Period 2. We're evaluating how to expand the independent spent fuel storage installation, preparing for the transfer of used fuel from the spent fuel pools to the ISFSI.

This period for spent fool management runs from 2014 to 2019. Our intent is to safely move spent fuel out of the spent fuel pools to the dry fuel storage pad in a safe and timely manner. We intend to finish that by mid-2019.

The following period you see, 2019 to 2031. This is an extended storage period where all the spent fuel is in dry fuel storage on the independent spent fuel storage pad.

Period 4, we're expecting the Department of Energy, at some point, to start taking fuel off our site and other sites around the country. Based on the current information we have from the Department of Energy, the old Unit 1 fuel, which is already partially

decommissioning, and all the Unit I fuel is already in the spent fuel system, would be removed first, starting roughly in 2031 through a 2035 time period.

And then the Unit 2 and 3 fuel, with what we currently understand the Department of Energy timing would be, would be removed from 2035 to 2049.

This is part of a bigger nationwide plan that the Department of Energy has to remove fuel from all the reactor sites, operating sites and decommissioning sites.

This is where we understand we would be in their queue to remove fuel from San Onofre.

And then ultimately, 2049 to 2051, with all the fuel offsite, we would then decontaminate and demolish and remove the independent spent fuel storage installation and go through a second license termination period related to the spent fuel storage system.

Site restoration, so think about this as now the non-radiological decommissioning. Again, we're in the planning period. You see a transition to site restoration.

We're actually -- the mesa is property on the other side of I-5, not part of the NRC's Part 50 license where we have warehouses and storage areas and

a training building, we're actually removing some buildings there presently. That's not a decommissioning activity, it's part of turning the facility back to the Navy.

You then see Period 3, and this is probably more operative for the power plant itself -- 2019 to 2024, you know, consistent with moving fuel out of the spent fuel pools, decontaminating systems, major component removal, we start the engineering for the sub-surface removal, engineering and permitting at that point.

And actual sub-surface structure removal is in the time frame 2028 to 2031. This will then follow the major component removal, the major building demolition, and then continue with sub-surface structure removal.

I think it's important to note at this point, San Onofre is on land owned by the Department of the Navy. So the power plant itself is under what's called an easement dating from the 1960s which granted the owners the right to build and operate a power plant on the facility.

So as we talk about completing decommissioning, we're committed to unrestricted, you know, release for unrestricted use, and you've heard

some of what that implies earlier, but ultimately the end use will be determined by the Department of the Navy.

So we have some work to do with the Department of the Navy to finalize exactly what their end use is, exactly what condition they would like to leave the -- use to leave the site in.

For example, most facilities that have been decommissioned don't necessarily remove every below-grade structure once the radiological decommissioning is complete.

So that's something that is up to the Department of the Navy. When I talk about the cost estimate, I'll tell you we've conservatively assumed we remove it all until they specify differently.

So this is what the completion then of the site restoration -- you see at the far end the ISFSI demolition.

Again, after the fuel is removed from the site by the Department of Energy, the remaining step will be the radiological clean-up of the spent fuel storage installation, followed by the demolition, and then a second licence-termination period.

That's a brief summary of what's in the Post-Shutdown Decommissioning Activities Report, I really encourage you go to songscommunity.com and take

a look at it.

We've shared it extensively with the Community Engagement Panel and other interested members of the public, but it's available to everybody.

Our Irradiated Fuel Management Plan, it was mentioned that this is not a PSDAR document itself, although it is summarized in the PSDAR and the Cost Estimate. But it's another important submittal.

Excuse me.

So the current dry fuel storage situation in San Onofre, we already have 50 canisters loaded with fuel. As was mentioned earlier, we have the horizontal NUHOMS systems by AREVA TRANSNUCLEAR -- 50 canisters of fuel, I have one canister of what's called greater than Class C waste from the Unit 1 decommissioning. This is the internals of the reactor vessel.

On the upper left, you see a picture of the site, and to the upper left of that, you see a rectangular box. That's where the facility is located.

And then on the lower right you see a close-up of the above-ground horizontal system that is currently in use.

Our current state on spent fuel storage, on the left side, we have 2,668 fuel assemblies in the two spent fuel pools, roughly 50/50 between Unit 2 and Unit

3, right.

Below that, you see existing dry fuel storage pad, it has Unit 1, 2, and 3 fuel in 50 canisters, 1,187 assemblies.

What has to happen, the 2,668 assemblies in the pools called wet storage will be transferred into approximately 100 canisters. The exact number of canisters will be determined once we finalize our decision on which canister design to go with.

Those of you who have been at the Community Engagement Panel meetings know that we're looking at a couple alternatives.

That's an approximate number of canisters, the number of fuel assemblies is accurate. It also includes 1,115 high burnup assemblies, and you've just heard a bit about high burnup fuel for storage and transportation.

So ultimately, we will have roughly 125 to 150 canisters, again, depending on the final design, on the independent spent fuel storage pad by mid-2019, awaiting removal by DOE offsite.

The Irradiated Fuel Management Plan itself is required by 10 CFR 50.54(bb). Again, there is a specific NRC format, specific NRC content guidelines.

It provides a written notification

discussing the plan that's required within two years following permanent cessation of operation, we filed that September 23.

It describes the program by which we will manage fuel and funding until the Department of Energy picks it up.

Our program is wet storage until 2019, followed by dry storage.

I've already given you these numbers -- 2,668 assemblies in the pools, an assumed start date of Department of Energy taking fuel from the industry of 2024, our target removal date from all our fuel off site is 2049.

We demonstrate that we have adequate funds already collected and invested in our trust fund for spent fuel management for this period.

And it also describes how we're going to maintain the spent fuel pools while they are in service.

Moving on to the third NRC document we submitted, it's the Site-Specific Decommissioning Cost Estimate.

And you've heard this alluded to, so I just want to summarize this.

Again, it's a required filing. This needs to demonstrate that we have adequate funding,

principally for radiological decommissioning.

Think of this as a companion, then, with the Irradiated Fuel Management Plan that discusses adequate funding for spent fuel management.

It also discusses the funding we have available for the non-radiological decommissioning. The NRC's focus is radiological decommissioning, and separately, spent fuel management. Our focus is that as well as site restoration.

So the plan discusses the decommissioning plan, describes funding, breaks down costs by the periods that I just showed you briefly, summarizes the things like the cost of services, undistributed cost, and outlines the cost for license termination, spent fuel management, and site restoration.

There are some key assumptions. We have developed a very conservative decommissioning cost estimate.

And I just want to highlight a couple things, for example, substructure excavation. This gets back to what the Department of the Navy will require.

For the cost estimate, we have assumed all the structure has to come out. So we've estimated the cost to assure we have adequate funds if the Navy would

like us to remove all the structures in accordance with the amendment, the current easement, we have adequate money to do that.

We've assumed the ocean conduits, these are the large pipes that bring intake water and discharge water from the plant into the ocean, that they are removed, and the cost of that is in the cost estimate.

For the Unit 1 decommissioning, the State Lands Commission approved a plan to abandon the horizontal conduits in place as more -- a more preferable alternative, as opposed to removing them, form an environmental standpoint.

I expect we will have the same treatment for Unit 2 and 3, from a more preferable alternative, but I have not assumed that for cost purposes. I have assumed the cost of removing them completely.

And you see some other things in terms of low level waste burial escalation, cost -- consumer price index for the D&D period followed by a higher escalation rate until we demolish the ISFSI.

So the real picture here, in 2014 dollars, the total cost estimate down at the lower right is about $4.4 billion, to decommission San Onofre completely, manage spent fuel through 2052 and decommission the independent spent fuel storage installation, and

remediate the site in accordance with the current direction of the Department of the Navy.

You see the breakdown. The green is actually the radiological decommissioning portion, $2.1 billion.

Down at the lower left, the yellow would be $1.276 billion. That is spent fuel management through 2052.

And then $1 billion for site restoration, the non-radiological cleanup of the site and removal of buildings.

As I mentioned in the stewardship principle, the trust fund is really rate payer money that we have collected and we have invested, and through prudent investment, the fund has grown significantly.

We are adequately and fully funded today. Oversight of the trust fund -- it's overseen by a five member committee, two of whom are internal to company, three external, nominated by management, confirmed by the Edison board, and the three external members approved by the PUC.

So the fund has good oversight with independence from the PUC, from a regulatory oversight. The NRC, the second bullet, regulates all radiological decommissioning, and the NRC regulates used fuel

management.

I cannot -- I am limited as to what I can spend on the decommissioning fund from an NRC perspective until the documents we have just filed are reviewed, the 90 day period is elapsed to allow them time to review and comment on it.

However, probably more fundamentally, the Public Utility Commission regulates the fund closely. So before monies can be withdrawn out of the fund and used for decommissioning, not only do we satisfy the NRC requirements, we have a Public Utilities Commission process to satisfy as well.

And both groups, the NRC and the PUC, will be involved in an ongoing basis in the -- ensuring the fund remains adequate, and then the PUC, making sure the fund is spent prudently.

In terms of open communication, public engagement, and education, there is a lot of things we're doing -- I have mentioned the Community Engagement Panel, and everything I have just covered, we have covered in multiple Community Engagement Panel meetings, and it has been posted on the website.

We really want this information available to the public, we really welcome questions.

We have held decommissioning education

fairs. We have had one in the local area. We've got another one coming up in the evening, where members of the public can come in and we talk about decommissioning.

We have public walking tours. We have now held two tours, open to the public to sign up, typically ten people.

We welcome anybody to come walk through the plant. We stay outside of the protected area, but you get a good view of the plant, and our staff explains our decommissioning plan and the decommissioning process.

Private tours for key stakeholders, be they elected officials or other interested parties, as well as a robust website with all this information and more -- and again, we want to make sure we're doing everything we can to communicate openly and listen actively.

And with that, I'll close and turn it back to the NRC.

FACILITATOR CAMERON: Thank you, Tom.

And there's a lot of information, and we're going to go out to you now.

And there were some sign-in sheets when you came in. I am going to try to follow those, but you may have a question, you didn't sign up that you wanted to speak, but there is going to be a meeting summary

prepared by the NRC that will be on the website, and there will also be the webcast.

So let's hear from a few people, and then you've been sitting for a long time, so we'll take a short break then, but you've heard the Community Engagement Panel mentioned, and we have several members here from the Community Engagement Panel.

But I am going to go to Dan Stetson first, for some comments from the Panel.

MR. STETSON: Thank you very much. My name is Dan Stetson, and I am the secretary for the Community Engagement Panel.

I want to once again compliment SONGS, Southern California Edison, for putting it together. We've had quite a number of public meetings where you've been able to come and ask questions, and we welcome those.

We've even had continuing workshops at the Ocean Institute where I work on an ongoing basis.

A couple of us had the opportunity to come and meet this morning -- David Victor, who is chairman of the CEP, Tim Brown, myself, and we were able to meet with some members of the NRC, and I just wanted to share with you just a couple quick takeaways from that meeting.

Number one, a big takeaway for me was that the NRC is not going away. They are here with us every step of the way. There are eyes, there are ears, there are inspectors, and they're the ones that are really going to walk us through this entire process, through the decommissioning and then also through the continued storage of the fuel there on site.

So that process is going to be very important, and we are really looking to them to manage this.

Also, as I looked out in that room this morning, I couldn't help but be impressed with the depth of experience of the folks that are there.

Quite honestly, they are different than you and I. We think we all speak English, but so much of it comes across as Greek. But I want to thank them for trying to bring it to a level, even this evening, something that we can all really understand.

But no matter what, I think we're all really interested in one thing, and that's the ongoing safety for this storage.

We know that the fuel is going to be there for an extended period of time. We don't really know how long. We want it out of there as soon as possible.

But as Al was saying, none of our cars are

going to last for 100 years, and something is going to need to be done if something comes up, so all of us are interested, if the worst case happens, do we have the plan and the resources to take care of us?

All of us want to trust, but we need the NRC to validate our trust. Thank you.

FACILITATOR CAMERON: Thank you, Dan. Another organization that started very soon after the decision was made to shut the plants down was the Coalition to Decommission San Onofre, and we're going to go to Gene Stone.

MR. STONE: Hi. I am Gene Stone from Residents Organized for a Safe Environment. I am also a member of the California Edison CEP, and I am happy to do that and happy to be here tonight.

I would like to thank the NRC for hearing our comments today, and I am happy to see so many friends that are here to support this whole process and the environment.

I must say that I am not really happy because I no longer believe that California Edison is considering doing the state-of-the-art decommissioning that they promised at the first Community Engagement Panel, nor do I believe that the NRC will demand or require that of them, but, unfortunately, a more

standard approach to decommissioning.

The NRC should have a more proactive approach to California's PSDAR. The fact that the NRC does not approve or disapprove this minimalist approach to the safe storage of nuclear waste is very disappointing and alarming to me.

Going forward with a plan that uses canisters that were designed for short-term storage does not make sense. It seems that we would be better served if the NRC would take a stronger approach in leading the industry into developing a much more robust canister system with defense in depth, not just talking about defense in depth, but real action items that the public can see that will help us in any situation that might arise.

Thank you very much.

(Applause.)

FACILITATOR CAMERON: Okay, thank you. Okay. And we're going to try to get to you, as many of you, as possible.

And unfortunately, it is just going to be one bite at the apple, so to speak, okay.

And if you have a question, you may want -- we can give you a follow-up on that, and the NRC has also told me that besides the comments that are going to come

in, the written comments, that if you send a question into them, that they will try to answer that question.

So if you don't get a chance tonight, we'll do that. And let's go to Donna Gilmore. Donna?

MS. GILMORE: Hi. I have been studying the issue, and what Gene said is correct.

We do not have a defense in depth system. We have 5/8s inch thick stainless steel that's the only thing keeping us from having a radiological accident that could result in us evacuating.

The canisters are subject to something called stress corrosion cracking. We do not know, of any of the canisters that are currently at San Onofre, we don't know if they have corrosion on them, we don't know if they have stress corrosion cracking, because they haven't looked at any of them, it's too dangerous for the workers to do that.

The canisters cannot be repaired. We're looking at potentially replacing canisters. There's no money in the decommissioning fund for replacing canisters.

There are canisters used in Europe and other countries that are much thicker, up to 20 inches thick, that were designed for maintenance, that were designed to be able to replace them, when and if they

wear out, and the ones we have are welded, weren't designed being opened.

Their part of the plan is to get rid of the pools. If we needed to replace a canister, we need the pools, but the NRC plans to allow them to get rid of them.

So yeah, I am not happy with this plan. I have got a website, sanonofresafety.org, and I have been keeping track of this. Everything is resourced with technical documents.

There was a lot of misinformation in the presentations, a lot of half-truths in the presentations, so I encourage you to check the sanonofresafety website to learn the part that was left out of this meeting. Thank you.

(Applause.)

FACILITATOR CAMERON: Thank you, thank you Donna. Let's go to Rochelle, Rochelle Becker.

MS. BECKER: I'd like to frame my comments on several NRC regulations that you pointed out that you're going to follow in this process.

The problem with you pointing out that you have NRC regulations, policies, processes, is a recent OIG report that said that didn't work when you replaced the steam generators.

So I have no idea why we're supposed to

trust you with the back end of the nuclear industry.

You didn't do the job right. You didn't require a license amendment request. We are sitting in the Public Utilities Commission trying to decide how much Edison gets, how much rate payers get back.

Rate payers are disgusted with this process. When the NRC fails, it is my wallet that this money comes out of, not yours.

When Edison fails, they try to argue that shareholders deserve more money than rate payers. Rate payers have very little input in this process. Rate payers pay virtually every penny of this process.

We are very tired of you telling us you have a policy, a procedure, a process that works. You don't. And we don't know that until it fails, and when it fails, we pay, and we're tired of doing so.

(Applause.)

FACILITATOR CAMERON: And Marnie (phonetic)?

MS. GLICKMAN: Thank you, I don't even know where to begin.

I have the Environmental Impact Report here that hasn't even been talked about, that anyone who reads it knows that it has to be old information. It can't possibly be what we know today about the hazards

for San Onofre becoming a long-term nuclear waste dump.

I am asking so much that you, both in Edison and our Nuclear Regulatory Commission, actually get involved with not passing the buck, not saying it's not our fault, we just have to stick to what we have as our possibilities.

I see that the regulation on the re-licensing these casks that now can't even possibly last with salt corrosion on the ocean as long as you're going to allow them to stay there, even if it's the minimum of 40 years.

It's not until spring that your 1927 regulation is going to talk about how we even take care of the process of checking these. We know now we don't have a way to.

I am asking for leadership at all levels and changes of the law. We can't leave that fuel on the ocean. It has to be moved. The fact that this plan is backwards, it is decommissioning the contamination that most of us could survive first, with our $2 billion.

They've changed even the term so that it seems like it's more important. None of us care if those domes go away.

We've gotten used to seeing them. We're not going near them. Leave them and leave the money in

the rate payers trust fund until all the fuel is off of location, somewhere away from the ocean. It is dangerous every day.

We have both security that I ask you to look for. You say you can't approve, but you do get to choose, you get to give waivers. Don't reduce the security.

Force an inspection of the 50 casks already there. Don't change the rate payer trust fund priority. Leave all $4 billion until all the fuel is gone.

And please, make sure that we don't lose priority in the government, DOE, coming to get our fuel by some way of Edison and the DOE changing the priority.

We need your help. We need our legislators' help. Let's get it moved to a place in the desert, away from the ocean, having the DoD help us get it there now, tomorrow. Thank you.

(Applause.)

FACILITATOR CAMERON: Thank you, Marnie.

We're going to go to Cathy Allen.

MS. ALLEN: Good evening. My name is Cathy Allen. I work at Age Well Senior Services in south Orange County.

We serve approximately 400,000 seniors

annually in south Orange County, and I have -- on behalf of them, I have a two part question for the NRC.

What is the NRC's process over time that there are enough funds available to maintain the dry cask storage on site indefinitely until the DOE resolves the regulatory issue?

What frequency do you perform an analysis, and what is the level of effort?

Thank you.

FACILITATOR CAMERON: When you say "level of effort," can you just explain that a little bit?

MS. ALLEN: The level of effort for your analysis.

FACILITATOR CAMERON: Okay. Do you understand the question, Larry?

MR. CAMPER: I was just going to say, is it financial analysis?

FACILITATOR CAMERON: Okay. So we're talking about finances. Can we -- who wants to give a -- okay, this is Mike Dusaniwskyj. Mike?

MR. DUSANIWSKYJ: Good evening. My name is Michael Dusaniwskyj. I am an economist with the Nuclear Regulatory Commission, and I am charged with making sure that the kind of question that the last question raised is my responsibility and my team's

responsibility, to ascertain the reasonableness that there will be enough funds to decommission and to take care of spent nuclear fuel for the foreseeable future.

The point that must be remembered is that the spent fuel is really the property of the Department of Energy.

Once it leaves the San Onofre Nuclear Generating Station's footprint, it is no longer in the custody of the licensee.

Until that time, there is enough funds to take care of the foreseeable future. Now, as far as what the lady is asking, is what happens beyond a certain period of time, I will be frank and honest with you.

Whatever possibilities may happen into the future, which all of us can postulate their own possibilities, we have to reason only with what is reasonable to what is required by the regulations, and what the Department of Energy's foreseeable future is in stake.

I am going to have to say something you are not going to like, and that is the fact that if you postulate some possibilities that the funds do run out, the solutions will not be popular. The financial solutions will not be popular.

But the point that must be remembered is

that the NRC does not regulate commerce. That is under the jurisdiction of the state Public Utility Commission, and it is our responsibility to make sure that all activities are done safely and completely.

And we recognize that safety takes money. Therefore, the NRC will not compromise on any level the safety of taking care of the decommissioning and the spent fuel nuclear -- excuse me, the spent fuel that's on site.

As far as how often we do the analysis, by regulation the licensees have to submit to us certain financial information that we look at to make sure that the foreseeable forecast is reasonable.

We want to make sure that the money is used exclusively for decommissioning. We check this once a year.

We also check on how much money is left. We keep checking on this until the license is terminated.

FACILITATOR CAMERON: Okay, thank you. Thank you for that question.

I am going to go to this gentleman here, this gentleman here, and then to Ace and Sharon Hoffman, and then we're going to take a break and come back. Yes, sir.

PARTICIPANT: Thank you. In regards to

the economic analysis you're performing, I want to know if the NRC, and you specifically sir that just answered the question, have you taken into account that recently at Diablo Canyon, you found the conditions for stress corrosion cracking after only two years in the task?

And you certified these, the NRC certified these, that they would be good for 30 years.

Now, my guess is, sir, that that two years takes your analysis and puts it on its head. And I want to know if you've considered that.

Furthermore, Tom, you've got some stones, talking about the California Public Utility Commission with all the news in the paper today about how they're under investigation, and how they're submitting all their paperwork. Really?

Now, as far as the car analysis goes, the casks that you guys are considering won't crack for -- excuse me, the casks that you won't consider won't crack in a marine environment.

Again, the casks that you guys are refusing to consider have the ability to be repaired. Everybody wants to repair their car. We also want the casks to be repaired.

The casks that you are not considering do not have the ability to inspect the exterior of the

canister.

You're saying that the industry has to come up with the solution in five years, but the NRC has given time and time again extensions to these kinds of rules.

We want it done now before it is approved. You guys, or the NRC, excuse me -- the casks that you guys are not considering do not have an early warning system before a radiation leak, i.e., you have no oil light for your car.

That's a major problem. You're expecting the public to accept the fact that you're going to go around and kick the tires on the car to ensure that there's some sort of integrity. That's a joke.

FACILITATOR CAMERON: I am going to have to ask you to finish up, sir.

PARTICIPANT: Thank you.

(Applause.)

FACILITATOR CAMERON: All right. Let me -- is Val (phonetic) here?

Oh, good, Val tries to keep control of Local 89.

MR. MACEDO: Thank you, sir.

Yes, that is not a hard task to do, is take care of Local 89. Local 89 has been taking care of the power plant for many years.

Local 89 was involved in the process since day one. Unit 1, Unit 2 -- decommissioning of Unit 1 as well, and then the building before the decommissioning of Unit 1 of the Units 2 and 3.

With that being said, I just want to start off by saying thank you for allowing me to be a part of the Community Engagement Panel, it's been an education for me to better represent my members.

I hate to use the word my members -- Local 89's membership, because I've heard different opinions from both sides, and I respect everybody, and I've heard a lot of safety concerns come into play.

My background, in the early 90s I was involved in the decommissioning of a uranium enriched -- U-235 enriched uranium processing plant in Sorrento Valley.

Of course, NRC was involved, Bechtel was the contract at the time.

So I'm very familiar with the decommissioning process at that scale. This is a much larger scale. I am thankful to be a part of the CEP panel.

I want to disclose something here. For years, for years, Southern California Edison has a huge amount of respect for me, for the safety that has been

put in place for our membership, for allowing me to be a part of the CEP panel -- it goes beyond that, as far as safety goes.

Local 89 has engaged in training. A lot of the members here, that are here today, don't have orange shirts on. There is no need to try to control the membership.

They are honest people. They are rate payers that built the power plant, that have the same voice as other people, and we're going to continue to respect everybody.

Those members that have been laid off, lost their jobs, because of the power plant going down, it's understandable.

But make no mistake, these members have gone through a serious curriculum in terms of training, on a daily basis, we have a large training facility, we have mobile sites, for not only Southern California Edison San Onofre decommissioning, but other contractors and other sectors of the market as well.

And I just want to say that I'm very thankful for the professionals to put all the information together, for allowing us to be a part of the process.

And let's give these members back their

jobs. These professionals are disclosing full-on information that's beneficial to the safety and the decommissioning of this project, and I ask that you vote and get this thing going, and let's put these people back to work.

And let's continue to have SCE allow whichever vendor, whichever contractor steps up to the plate and becomes the awarding body for the decommissioning project to follow the safety process that it's had for years and make sure that all of our members, and anybody else that joins whichever union that goes into the decommissioning of this project, that allows them to go home safe to their families.

I thank you for the opportunity to speak.

(Applause.)

FACILITATOR CAMERON: Thank you, Val.

And thank you very much, and we're going to hear from Gary Headrick. We're going to go to Ace and Sharon. And we're going to take a break quick for ten minutes, and we're going to keep going. Gary?

MR. HEADRICK: Thank you. Can you hear me? There we go. My name is Gary Headrick, founder of San Clemente Green, and we've been concerned about our safety ever since being contacted by some of the great workers at San Onofre, who we have a lot of respect and

concern for.

Just one note about Val, I -- you know, we had a conversation shortly after the plant was announced to be shut down, and Val was concerned that the NR -- that he was being reassigned to fight fires when he hadn't been trained to do that in a radiological environment, but I think he's since been trained, so I am glad that's happened.

But the point I want to make about what I've heard tonight is, actually, the most encouraging words I've heard is that you're trying to keep ahead of this.

You are recognizing that you are not ahead of this issue about how we are going to deal with the nuclear waste on site, and deal with it safely, and I appreciate that.

But the problem is, we're still rushing ahead to this -- expedite this decommissioning thing, which we all want to see happen as quick as possible, but we all agree that it has to be as safe as possible.

And I believe that you need to seriously look at this problem longer, and not play word games about whether you approve it or not.

You are the regulators. You make the regulations. And you're also guilty of making exemptions for regulations whenever the industry wants

it, and that's a real problem you have to get over if you want to earn the public's trust.

So I'm appealing to Tom Palmisano, who I appreciate and respect in depth. I think Edison is the one who we can thank for having shut down the nuclear power plant and done the responsible thing when the NRC did not step in and do -- prevent them from restarting a broken reactor without fixing it first.

You know, so Tom, please, give us some time. Let's look at that more sturdy cask system. Let's not rush into this, regardless of what the NRC decides.

We need you to do the right thing, and we're putting a lot of faith in you. We've got to get this right. Thank you.

(Applause.)

FACILITATOR CAMERON: Thanks, Gary.

And Ace, why don't you come here so that they can get you on the camera? Okay, go ahead.

MR. HOFFMAN: I agree that the people who worked at that plant were good all along.

I've been fighting them for 20 years. I've never had a problem at any of these meetings. Nobody has ever come up to me and said anything nasty or mean. I think you're all good people.

And if we decommission that plant right

now, one of them is going to die, on average, because of the difference in the amount of radiation that they're going to be exposed to. And I wonder if they're ready to pick which one of them it's going to be, or whether it's going to be a multitude of them that are going to get some kind of cancer.

Now that is what is going to happen, because the NRC's own statistics say that there's going to be a couple hundred rem dosage if we do it now, and it's only going to be a couple dozen rem if we do it later.

So I'd like to see us wait. But what I'd really like to talk about is that 5/8ths inch thick protection against a -- that's the radiological barrier for the next 60 years, is that 5/8ths inch stainless steel.

The 3 to 5 inches of concrete is virtually irrelevant because it's designed with vents, so if something gets in there and damages the casks, we are going to have a problem.

ISIS is training six year old kids to be suicidal terrorists. ISIS is interested in a multi-generational attack against America. ISIS is producing videos in which they are recruiting people in order to attack America.

Now, Tom, I am sure that you are aware that

there are many, many weapons that can go through 5/8ths inches of steel. This is a multitude of them.

We have to protect that plant better. We can't pretend that this -- that just because we're no longer a nuclear power plant, that we're no longer a target.

We are most definitely a target because of where we are and who we are, and because of our freedoms, and because we can have a meeting like this. This makes us vulnerable.

So you have to consider not just the things that they tell you to consider, but everything that your heart tells you to consider.

That's really all I have to say, this is practically the last meeting that we're going to need to attend here.

And I want to thank all of the workers at San Onofre. I don't believe any one of them did a bad job.

(Applause.)

FACILITATOR CAMERON: Thanks Ace, and last commenter before the break is Sharon.

MS. HOFFMAN: Good evening everybody, and thank you for giving me an opportunity to speak tonight.

I have a few questions.

The first question is really kind of simple. Does anybody have the ability to say no? In a business environment, there's a big difference between the people who can say yes and the people who can say no.

And I didn't hear anybody from the NRC say I am the guy, I can say no, I can say stop, I can say we need more information.

And I think that's important. Somebody should be able to say no.

I also want to say to the gentleman who spoke about the finances that realistically, assuming that the DOE is going to take all the fuel from San Onofre by 2052 is not realistic.

If you look back to 1988, say, we would have said all that fuel that has already been accumulated would be gone by now. And it's not. So let's be more realistic than that.

And last but not least, let's not be in a rush to put this fuel into casks. It -- keeping it in the pools gives us an opportunity to observe it, gives us -- gives SCE and the NRC the opportunity to learn something about the aging of casks.

We're talking about leaving fuel in casks for hundreds if not thousands of years, and we have no

experience in that. So what's the rush? Thank you.

(Applause.)

FACILITATOR CAMERON: Okay, thank you, Sharon. Let's come back around, let's come back around five to 8:00, that's like seven or eight minutes, and we'll get started down here, okay, and we'll keep going. Thank you.

(Whereupon, the meeting went off the record at 7:48 and resumed shortly thereafter.)

FACILITATOR CAMERON: All right. Are you ready, are you ready to go? And please introduce yourself.

MS. JOHNSTON: Hello. My name is Chris Johnston. I thank you all for coming, and I want to believe that your safety recommendation is number one.

And I'm concerned about our community, I'm concerned about eight and a half million people.

I would not want to be sitting in your shoes. You have a very grave responsibility before you.

I have concerns about the canisters, as other people have expressed -- the NUHOMS canister that's 5/8ths inch thick, that you have done some salt spray testing on at Diablo Canyon, and found stress corrosion cracking.

And we are talking about atmospheric corrosion here at San Onofre. So deeply concerned that these casks, these canisters, are not going to withstand atmospheric pressure.

And beyond that, I'm also wondering, and I believe this is a question for Al, regarding the seismic rating for cracked canisters. Can you answer that?

MR. CSONTOS: Can you hear me?

MS. JOHNSTON: Yes.

MR. CSONTOS: All right, first of all, let me get to your -- one point you mentioned about seeing stress corrosion cracking at Diablo, it has not happened, okay?

MS. JOHNSTON: It has the potentiality to happen.

MR. CSONTOS: There are chlorides there, okay?

MS. JOHNSTON: Yes.

MR. CSONTOS: There were chlorides that were found at Calvert Cliffs, there were chlorides that were found at Hope Creek, as well.

They were on brackish water. Diablo is right there, that's the coast.

That's one of the reasons why the EPRI went ahead and did that analysis, or went ahead and had

volunteer sites -- these three were volunteers, okay. They didn't need to do it, they weren't required to do it, they just volunteered to do it, okay?

So that showed a lot of stewardship on their part, to do that.

Now, about two weeks ago -- all right, so about July, we held a public meeting. We went out there and showed our cards, and we said, this is the Aging Management Program that we want for stainless steel canisters, period, okay?

That required inspections, that required operational experience evaluations, that inquired -- that required corrective actions, quality assurance, all that, okay?

Two, let's see here, about -- that meeting was, after that meeting there was another meeting on chloride stress corrosion cracking. And I know several folks were on that phone call, okay?

That meeting went very long. We were kicked out of the room because we had so many questions, which is great.

I talked to Donna after that meeting about these issues. That meeting was the prelude to the report that EPRI was about to provide. Now that report was published two weeks ago.

MS. JOHNSTON: I read it, yes.

MR. CSONTOS: No, that's a different one, I think it -- the Flaw Evaluation and Growth Calculations, have you heard of that one?

MS. JOHNSTON: No.

MR. CSONTOS: Let me give that to you. You can go to the EPRI's website, and I am going to pull that up here.

Oh, it's back here. It's report number 3002002785. Do you want me to repeat that?

MS. JOHNSTON: Yes. I, I, yes, it's okay, I will be watching the -- she's got it.

MR. CSONTOS: So -- so in that report they did a calculation, a much more sophisticated calculation than what we had intended to do, where we just did a quick calc, okay, for Calvert in specific.

Now what they evaluated is they looked at all the conditions, they went to the Camp Pendleton -- is it Pendleton? -- Pendleton site here, and they pulled out all the information on the atmospherics, the weather, and everything.

Pulled out some of the calcs that they did, were very, were very sophisticated based on that.

Their worst case scenario, okay, this is -- there's two, you've got to remember, there's two, two

-- two stages of stress corrosion cracking for chlorides.

There's the incubation period, where you get all the environments, you get all the conditions that could start cracking, okay.

Then you have the actual crack growth. Okay, there's two stages -- incubation plus initiation, and then crack growth.

They just only looked at crack growth. They did not look at how long it would take to get to that stage, because just having chlorides there is not sufficient to have cracking to start, okay?

You need to be in a relative humidity environment, you need to have the material conditions, you need to have a lot of things that have to play out to get this to start cracking.

We raised this issue nearly nine, ten years ago. We're at the stage now where we're trying to act on it.

The calcs that came out, okay, and I'm going to give these numbers to you -- for ambient plus 15 degrees, that means 70 degrees plus 15 degrees Celsius, so that's about what, 20, 25 degrees Fahrenheit, roughly -- to get a 50 percent through-wall crack, this is just not assuming how long it takes to start, to get the

incubation period to begin, that could take years to decades, okay -- you have 18.2 years, okay, by their calcs, to go through-wall of a half-inch thick canister.

To go through-wall, completely through-wall, it's 60 years, over 60 years, okay. To go 75 percent through-wall it's about 40 years.

Okay, now if you take it to the 5/8ths inch canister, you're talking about, worst case now, worst case, 28 years to go halfway through wall, and to go full through wall, over 86 years.

So I want you to understand that this is the -- we're, we're trying to get ahead of this. That is what I am trying to say.

MS. JOHNSTON: What is your defense in depth, in that case, if it does happen?

MR. CSONTOS: If it does happen, you know, this is why we're doing the inspections, okay?

MS. JOHNSTON: How can you inspect canisters that can't be inspected?

MR. CSONTOS: Well, you mentioned that they already did inspections at Diablo, and I mentioned Hope Creek, and I mentioned the volunteer inspections there.

There are also Calvert Cliffs, the inspection for their license renewal, Oconee and Surry

did as well. Those are visual exams --

MS. JOHNSTON: Okay, you're saying you can go all around the canisters and you can see everything, every potential crack?

MR. CSONTOS: So -- part of what, well, what we do here, and that's where in the meeting that we had that was back in that July time frame, we talked about a Koeberg plant, that was a South African plant, Donna, you showed me that slide that I showed you earlier, or that we had at that meeting.

The Koeberg plant had cracking in a 304 tank and a pipe, okay?

And they went ahead and inspected, and they repaired it. So there are repair technologies, and we have seen inspection technologies that are out there, okay?

MS. JOHNSTON: Okay, now back to the final question, is the seismic -- the seismic rating for cracked canisters back to --

MR. CSONTOS: So we did a calc on that --

FACILITATOR CAMERON: Thank you, Chris. Seismic.

MR. CSONTOS: -- so let me talk about the calc on seismic, all right.

What San Onofre did is they analyzed to six

G's, which is ten times what their design basis was for for this 1029 system.

We went ahead and analyzed for the loads that would occur for the six G's, so that's ten times what a earthquake would be here for, what the design-basis earthquake, okay, ten times that.

You would have to lose over 80 percent of the entire canister thickness, the entire canister thickness, before you would have any issues, okay, for a ten time seismic.

That's the robustness in this design. That is what I am trying to get across, that this design and these systems are so over-designed for these types of conditions.

Now, yes, you know, we don't want to ever get to a crack that is 75 percent through-wall.

That's why we're doing this inspection. That's why we're doing these aging management plans. That's why we're trying to get ahead of this, okay?

The first -- the renewal for San Onofre will be in 2023, okay? 2023 is right around the corner.

At that time, we will inspect, probably the worst -- the worst, in terms of our environmental, our, you know, evaluation.

EPRI, that's the Electric Power Research

Institute, that did this Flaw Evaluation Study, is doing a compendium that's coming out this year sometime.

That's looking at the environmental factors, the prioritization. And from that, we're going to prioritize which canisters, how we can focus our inspections on a specific canister, okay?

At that time, that's what we're going to be doing for -- by the time that San Onofre comes around, this technology will have been used.

We have put into the Calvert Cliffs renewal, we have put in there what they need to be able to inspect, okay, within the three years.

FACILITATOR CAMERON: Okay, thank you. And we did get an extension until 20 after 9:00, okay, so we have some more time, but one other suggestion is, and this all depends on Al's goodwill and schedule and all that stuff, but we need to be out of this room by 20 after 9:00, but there's no reason why you can't have an informal conference with Al out there somewhere, okay?

So I am glad you are not listening, Al, because I got you committed. Okay, great.

Let's go to Kevin, Kevin Blanch, back here, and then we're going to spend some time here, go to Richard McPherson, we'll go over that side of the room.

Kevin, go ahead.

MR. BLANCH: Well, I want to first say that the NRC, which I call you guys the Nuclear Rallying Cheerleaders, I don't want any of you to take this personal, what I'm about to say.

I would be more than happy, I am staying at the hotel, to sit down and have a beer with any one of you, and I'll even buy, because we always buy.

I have been fighting San Onofre since I was a little boy. I was given two months to live three years ago. My father was nuked to death in the Nevada test site from Pendleton. I have been anti-nuclear.

I want to ask this simple question to the NRC, and again, do not take this personally, I call you the Nuclear Rallying Cheerleaders.

You have no Congressional powers whatsoever, none. You are the Nuclear Regulatory Commission. You are not the executive branch, you are not the legislative branch, you are not the judicial branch, none of the above.

Congress makes law, not you. And when you come out and slide in and say oh, we're turning America into America Toxic. We are going to take every one of these catastrophes, which is the biggest mistake in human history called nuclearism and turn them into

official nuclear waste dump sites.

Only Congress has that ability to do that, not you. Not the NRC. I know Allison. I am glad Allison resigned, and hopefully on November 4th, maybe Harry Reid, who is protected from putting this monster in Yucca Mountain, which Congress was promised to put in Yucca Mountain.

I sat right here in the 60s and listened to these same ridiculous conversations about what we're going to do with the waste, where we're going to put it.

That stuff needs to come off that cliff yesterday. San Onofre is the poster-child for everything wrong with the nuclear industry. It is a catastrophe, and if it was up to the Nuclear Rallying Cheerleaders, it was all the grassrooters.

And I would like to say thank you, and get this into the public domain, to the grassroot activists in Southern California.

I live in Utah, which you parked the generator in my backyard, illegally.

I would like to say thanks to Gene, Ace, all these people that have fought this tooth and nail. It was us as grassrooters that exposed the lies that was going on by PG&E, the great crimes against humanity.

You protected them. The lie on the

exchange of the generator, these crimes against humanity that were taking place -- you are not Congress.

Yucca Mountain is a bad idea. Shooting it into space is a bad idea. But I'll tell you what's really a bad idea. Nuclear. Building a nuclear site on a cliff, on the most beautiful beach in the world, on a cliff, and then storing the waste there for 40 years when it's not built to dispose of the waste.

Dry cask is a catastrophe, it's a pathetic, stupid idea. Yes, Yucca Mountain is pathetic, but Congress passed Yucca Mountain. You're not Congress.

Ship it to Yucca Mountain. Yeah, I know that's a horrible debate and Congress can't get anything done, but we still have a Constitution. And it was approved and passed to be in Yucca Mountain decades ago.

And Harry Reid blocks it. Yeah, I understand. Born and raised at Yucca Mountain, he has to get elected, his power.

But again, don't take this, any of it, personally, but I consider the NRC to be nothing but a group of criminals who protect other criminals.

San Onofre are criminals, period, period. What they pulled on the exchange, the crimes against humanity -- nuclear fallout is leukemia. My father died of it, and now I'm fighting for my life with it.

So many are.

That stuff needs to come off that cliff before we get an earthquake. Dry cask, whatever you will -- not stay there. It is a catastrophe, ticking time bomb. Thank you.

(Applause.)

FACILITATOR CAMERON: Okay, Kevin.

We're going to go to Alan.

MR. WOO: My name is Alan Woo. I am with the Asian Pacific Planning Council of Orange County.

I used to be chair of the Low Income Oversight Board with the CPUC. So I understand a little about the rate payer and what regulatory agencies kind of look at.

I could kind of appreciate Southern California Edison though, in making a decision to stop operation.

They did that voluntary. There might have been some encouragement, but nevertheless, here we are, right? We're closed, we shut it down, we're trying to come up with a plan to contain it, store it, and then go forward as those policies that the federal government has to yet come into place.

Something has to be done now because I did a calculation when 2052 would be, that's 38 years from

now.

I may not be around, and maybe most of you guys are not going to be around, and most of the activists here won't be around too, but we leave in place what we start today.

And I could appreciate that you thought about inspections, you thought about containment, you thought about storage, you thought about all these things, and that each one of you pledged that you are going to become good stewards in terms of making sure of the future.

So at least now, I could see the faces of people who are committed to doing something to decommissioning it in a responsible way, to contain this situation that we have, and to move forward.

I don't know what's going to happen in 38, 40 years, who may be sitting up there, if there is anyone, whether anyone cares in Washington, or whatever.

But I know today, you know, there's people in the community that cares.

I know that we have a utility company that cares and has been transparent, and has been talking to us in the community about the impact on us, and you had a lot of workers here who need to go back to work who

know and care how to clean up this thing.

So that's all I'd like to say, I just think that you presented a large plan here, it covers every area including environmental impact, we just got to see, if we don't start now, and we postpone it, you know, you are just postponing a problem and you're not getting started.

FACILITATOR CAMERON: Thank you, Alan. Donia?

MS. MOORE: Good evening. My name is Donia Moore, I am with the San Clemente Chamber of Commerce.

I'd like to thank the NRC and Edison for having this forum and allowing us to speak here.

I am aware, of course, as you've discussed often tonight, that the NRC gave Edison 60 years to do the decommissioning process, and they voluntarily decided to do it on a 20 year timeline, which I think is very commendable.

And I understand the need to remove the waste that's being stored currently at San Onofre.

However, I have a question, and that is that one of the things that hasn't really come up much is that, as I understand it, it's actually the federal government that is the body that is giving permission

to remove the waste, and as where to put it, and there's no repository at this moment.

And I haven't heard anything about when there will be a repository to remove the waste to, so it's great to talk about removing the nuclear waste, but how are we going to do that if we don't have any place to store it?

And as I understand it, that's up to the federal government, is that correct?

FACILITATOR CAMERON: Should we go to Keith McConnell for a little bit of history on this? This is Keith McConnell, who is the Director of the Waste Confidence Directorate who did the Environmental Impact Statement and Rulemaking called Continued Storage.

But Keith, do you want to provide some information to Donia on that?

MR. MCCONNELL: Yes, it is the Department of Energy's responsibility to manage the disposition of spent nuclear fuel from all the power plants across the country.

The Department of Energy's most recent issuance in terms of a repository was put out in January of 2013.

What they envisioned at that time was a stage process where there would be a pilot effort to take

the spent fuel to a centralized interim storage facility, and then there would be a full scale interim storage fuel facility in the 2020, mid-2020s time frame.

And then a process where they would go through and solicit interest in development of a geological repository other than Yucca Mountain.

And that's all in this strategy that they issued in 2013. Now there is legislation required to allow that strategy to proceed, and it also depends on community involvement and state involvement in terms of looking for a repository.

But that repository would not be available until the 2048 time frame, and that ties into the presentation that San Onofre gave today.

Does that help?

MS. MOORE: Yes, thank you, it does.

FACILITATOR CAMERON: Thank you, Keith. And let's go to Heather and Steve, and then we're going to go to Richard, and then we'll go over to Ray and everybody over there. Heather?

MS. JOHNSTON: My name is Heather Johnston, I am the Executive Director of the Dana Point Chamber of Commerce.

And I am interested in more of the process, so my question is, is what will be the daily

responsibilities of the NRC resident inspector? How are their activities changed as the project progresses? And how are you going to communicate that to the community and to our local businesses?

MR. KELLAR: I think that's probably a question for me.

There is currently a senior resident on site. That person stayed on site after San Onofre had shut down.

Long-term, that position will not be there, day-to-day, but will be a regional-based inspector that will be in concert, or not in concert, but in contact with the licensee, and they will be aware of any of the higher risk activities as well as the activities that the licensee is undertaking.

So the resident inspector will not be there, but the inspector from the region, the teams and the individuals will be going out and performing the inspections, particularly based on the higher-risk activities.

Now, what I mean by that is, as the licensee prepares to move the fuel from the spent fuel pool to the ISFSI, that would be a higher-risk activity. Some of the dismantlement of the different systems that had radiation and radioactivity would also be a higher risk

activity.

So we would have inspectors that would be there and watch those higher risk activities, as well as, I tried to explain during the presentation, there are certain aspects of the day-to-day operations that also are inspected on a routine basis, such as the organization of the licensee, how they maintain the business, how they perform their audits, how they perform their self-evaluations.

All those are looked at, and they're planned in advance, and the inspection teams would go out and inspect those, as well as the higher-risk activities.

MS. JOHNSTON: Thank you.

MR. KELLAR: You're welcome.

FACILITATOR CAMERON: Okay. Is this Laurie (phonetic)? Okay, why don't we stop here. Go ahead.

PARTICIPANT: Hi. I would like to ask the question of this gentleman on the end here, okay.

The gentleman on the end here made a statement about dose rates, and you said that you compared it to background radiation in certain areas of the world.

And I just wanted to state that what's in

the casks there at San Onofre is not background radiation, it's the most dangerous manmade radionuclides in the world. And you can't compare that to background.

And if the rest of the NRC stands behind that statement, then that's a criminal act.

FACILITATOR CAMERON: And you're talking about Mr. Camper on the end? Larry, do you want to clarify anything on that one?

MR. CAMPER: Yes, I will. Thank you for the question.

What I was trying to do was to draw a comparison as to what it means in our standard that we require.

When that site is ultimately decommissioned, it has to meet that dose standard to be released, and that's 25 millirem and ALARA.

What I was trying to give some perspective on what is 25 millirem, because a lot of folks look at that and they say, well, what is 25 millirem?

So I was trying to draw a comparison. If you get on an airplane and you fly to New York, it's three millirem. The natural background radiation in the United States, as I said, is 300 to 600 millirem, depending on where you are.

I agree -- I beg your pardon?

PARTICIPANT: That's not massive radiation.

MR. CAMPER: Oh, I agree with that, I agree with your point fully. If you're saying that there's a dose -- I'm sorry?

PARTICIPANT: The problem at San Onofre is not the background radiation.

MR. CAMPER: Oh, I agree with you fully, it's not background radiation. Of course it's not.

FACILITATOR CAMERON: Okay. Sounds like there's some agreement here, but let's let Larry finish, and then we're going to go to some other people, thank you, Laurie. Larry?

MR. CAMPER: No, I agree with you that it's not background radiation. I was simply trying to draw some reference to what 25 millirem meant when that site is decommissioned to satisfy that dose standard.

FACILITATOR CAMERON: Okay. Steve?

MR. ADAMS: I am Steve Adams (phonetic), a resident in Laguna Niguel.

And reading through the report, I had a question, couple of questions.

The term "islanding" of the pool, the spent fuel pools, was used, and I'm not clear on what islanding

would be in those pools.

And then my next question is based on listening to some of the previous commentations. Is it safer for the spent fuel to be in pools, or is it safer for it to be in the casks that are proposed right now?

FACILITATOR CAMERON: Well those are good questions. And who would like to take the islanding on?

MR. PALMISANO: So let me take that on, because I think that came off our slides.

FACILITATOR CAMERON: Okay.

MR. PALMISANO: That is a term that refers to taking the current spent fuel pool cooling system which are installed plant equipment that will be taken out of service.

When we say "islanding," that means putting in dedicated standalone cooling systems for the spent fuel pools. So we essentially island it and separate it from, disconnect it from the rest of the installed plant equipment. That's what the term islanding means.

FACILITATOR CAMERON: Okay. And does anybody want to talk to the question that's been percolating for a while now about is it safer in pools or dry storage? Doug?

MR. BROADDUS: So, the -- our requirements are that both processes have to be safe, and so we

consider them, and we've looked at them in the past, you know, to see, is there benefits one way or the other?

And there was recent -- there was a recent study that was actually put up to the Commission, to our Commission, to see whether there was a need for their to be an expedited transfer of fuel from the spent fuel pools to dry cask storage.

And the determination was that there is no real benefit from that standpoint to do it in an expedited manner, so to get in there sooner.

In some cases, you can't actually take the fuel and put it into spent -- in dry cask storage, within -- until after a certain amount of time, because it just can't get into the casks, the casks can't handle the amount of heat that's going to be generated by that, so it has to stay in the spent fuel pools.

But the spent fuel pools are build to be very -- very rugged structures that are going to withstand the same seismic types of loads and such that the casks would have to be able to withstand as well, to be able to dissipate the heat in the same manner as the casks would, but just through a different process.

They're dissipating it through the water that they're in, whereas the casks are dissipating it through the transfer into the air, or the atmosphere

that's within the casks themselves. So it's a different process.

(Off mic question.)

FACILITATOR CAMERON: Let me -- let me just, this will be the last question, but go ahead.

MR. ADAMS: No, I was referring to the comment that was made about, like, a terrorist attack. Is it being in a pool, is that safer than being encased in stainless steel and concrete?

MR. BROADDUS: All right, so, from the standpoint of terrorist attacks, obviously with -- you can't quantify, it's very difficult to quantify what type of a terrorist attack has to occur.

So what we've tried to do, and what we've done, is establish what's called a design-basis threat, and that's a threat that has been established to provide assurance, very high assurance, of that they're going to be able to, the licensee would be able to thwart an adversary coming in to, and then attacking it.

So that's what the -- from a terrorist standpoint, our defense in depth is to ensure that they have a good strong security program to ensure that the terrorists are not going to be successful in whatever attack that they would do.

FACILITATOR CAMERON: Okay, thank you,

thank you for that. Richard McPherson.

MR. MCPHERSON: Good evening. I hear a lot of negative information, about 90 percent of the people are talking negatively about nuclear power, something they don't really know about.

50 years ago, I suited up in NICs (phonetic) for the first time, and I participated in a defueling of a reactor and a refueling of a reactor.

And I have been involved in that process ever since. I was even selected to represent the United States at the International Atomic Energy Agency for four years on something called Nuclear Fuel Cycle Facilities, which is the front end and the back end, the environment and public opinion.

And I have now been involved for almost 51 years in nuclear power. I have little or no concerns over what you are presenting here because I've seen it presented so many times, and I've seen it be successful.

It's unfortunate that there's so much misinformation out there and there's so much misinformation that's generated on purpose to cause fear in people.

But as an operator of five nuclear power plants, and I've been involved in refuelings and been involved in the back end of the fuel cycle since, a lot,

since 1992, I appreciate everything you are saying and doing and I think we're going to be safe.

Thank you very much.

(Applause.)

FACILITATOR CAMERON: Yes, sir.

MR. GARDNER: Yeah. God evening, I'm Richard Gardner (phonetic) from Capistrano Beach.

I feel that the process is in place to lead us to a safe condition, even though I think I agree with everyone that getting the spent fuel out of the, off the site and in a permanent repository would probably be the long-term benefit, and that's not in your hands.

What I am here to suggest is that the San Onofre Nuclear Generating Station should be re-purposed into a reverse osmosis drinking water facility, at least for the intake structures and the turbine buildings.

And I believe that, you know, without actually doing the, you know, preliminary design, that it would be easy to have 50 million gallons a day, or 100 million gallons a day.

And that water would be provided, and it would be adequate supplies for the Marine Corps that live on the base, and also it would satisfy the demands of at least 25 percent of south Orange County, which goes, which covers a great distance.

So I think that's a much higher and more important use, considering if this drought does not -- if we don't get some rain pretty soon, we could be drawing Lake Mead down, and we may be in a position where we would have hundreds of thousands, if not millions, of people dependent on a water source that isn't there.

So that's why I think it would be best if there was a planning, a planning facility where we would look at a contingency of converting or re-purposing the plant, and doing it in a way that the transition could occur, to whoever the third party is, and it could be Southern Cal Edison but it could be anyone else, and that the discussions with the Navy happen immediately so that we could save Southern Cal Edison over $130 million for the demo of the turbine buildings alone, and we would save the water authority another $100 million for the new facilities.

And you know, the plant here in Carlsbad is costing $1 billion for an RO plant to produce 50 million gallons a day.

So we could have that, and I think we should begin to focus on that's more important than many of the things that we are now concerned about that are hypothetical.

So if anyone can figure out how to get

through the right departments within the federal government and in the state, to think about how it affects Southern Cal Edison's planning process, I would like to be actively involved. Thank you.

FACILITATOR CAMERON: Okay, thank you. Let's go over here to, let's go to Ray. And then we'll go to --

MR. LUTZ: Okay, hello, my name is Ray Lutz, I am with citizensoversight.org.

I wanted to bring up some of the things that concern the public about the process and where we are today.

I am one of those that are very happy that the plant is shut down. I don't think that nuclear power is a good idea, I don't think that it's safe to have all this waste around.

That's the definition, the non-definition of green. If you have a whole bunch of waste at the end, it's not green energy.

And we have a whole bunch, 3.2 million pounds of highly radioactive waste that we've got to deal with now.

The problem is that it almost seems like we're seat-of-the-pants operation here. Almost nothing has been really planned ahead -- you say, we've

decommissioned all these other plants, but we really don't really have a solution for this fuel, still.

It's really not a good plan yet. And no one is sitting here saying, here is our plan for the next ten, twenty, thirty years. There is no plan. All we got is here. That's the end of the plan, a foreseeable future.

You say, for the foreseeable -- what the hell is that? What is a foreseeable future? Ten years? Is that it? Is that the foreseeable future for the NRC? 20 years? I mean, what is foreseeable?

Because right now the only plan is leave it here on the coast. Edison doesn't have a plan for the next ten years, the next place. Everybody is pointing fingers. The Department of Energy's responsibility, not us.

NRC, we don't even have to approve the damn plan. It's not our responsibility. If it goes south, eh, we didn't approve it. That was their problem, because we have a way out, we've covered our butts. We don't have to approve the plan.

Who came up with this? Who came up with the fact that the actual license amendment process is at the very end? When everything is done, we finally talk about whether we should approve it.

Like, in 50 years from now, we're going to talk about what we did was 50 years ago, was okay.

The NRC is all backwards. You have the license amendment for the termination at the very, very end, instead of doing it -- why did they do that? And it's because they said, oh, there's going to be some activists that are going to be out there and they're going to put blocks in what we're doing, so I'll tell you what, let's put the license termination plan at the very end, so that no one can block it.

That's where we are now. The NRC doesn't even have to approve it.

Seat-of-the-pants operation, no approval, no long-term or even 10, 20 year plan. These are all big problems in our minds.

The fact that the license termination is at the very end of the process. It should have been now. And you're saying we don't have to approve it.

My proposal is, you are going to have to work around this system. Stop pointing fingers at the Department of Energy.

Southern California Edison, thank you for shutting it down. Now you need to do your part by planning, here is what we are going to do with the fuel over the next twenty, thirty years. Make a plan. It's

not going to be perfect, but at least get it started.

NRC, do your part by approving this plan. I don't care if they say it's not your, you can't approve it -- do it anyway. Say, we are approving this, and you're putting your butt on the line.

And I expect you to be able to say we're not approving it, and Southern California Edison will cooperate. And they'll say, you know what, you guys haven't approved it yet, there are some problems? We're going to fix those. I know Tom Palmisano would love to do that. He would love to fix any problems that you've got.

So don't say, oh, it's the end of the clock -- we're not done yet, but go ahead. Make sure that everything is done, all the i's are dotted and all the t's are crossed.

I think it's right. You know, this thing about the fuel pools, the guy just asked, he's not here anymore. That study that you said, oh, it's the expediting the fuel out of the fuel pools into the dry casks -- the only reason they said that was because of cost.

That wasn't because of a safety issue, and the whole system that you guys use at the NRC is screwed up because you start with the cost analysis before

you've ever gotten done with the first, the safety analysis. They jump right away into the -- whether it's cost effective or not.

And so that's why they came up with that. It probably is better to put them in the dry casks. Let's face it. If you have a terrorist attack, the fuel pools are not as safe as a dry cask. And that should be the answer.

Sir, you answered that? Your answer should be, the fuel pools are not as safe as a dry cask in a terrorist attack.

You know, you've got to say something -- you think they're just equally as safe? Is there an answer?

MR. BROADDUS: I explained that it's the design-basis threat, and the licensee has to protect against the design-basis threat, and regardless of whether they are attacking the spent fuel pools or they're attacking the casks, they have to be able to thwart, they have to be able to thwart the adversary --

FACILITATOR CAMERON: Okay.

MR. BROADDUS: So it -- and prevent the attack from being successful --

FACILITATOR CAMERON: Thank you.

MR. BROADDUS: -- so it doesn't matter where they attack.

FACILITATOR CAMERON: All right. Ed.

DR. SIEGEL: Yeah, hi, my name is Dr. Edward Siegel (phonetic), a graduate of Berkeley, Michigan, Michigan State, MIT, Westinghouse Combustion Engineering GE, PSEG, and the IAEA, many years. Son of Sidney Siegel, co-director of Oak Ridge in the beginning.

What I heard today is summarized best by a Harry Belafonte lyric, "It was clear as mud but it covered the ground."

One of the problems that occurs is in this document, which I just got from someone here, which I presume is available, my goodness, people at Southwest Research Institute found that alloy composition dominates things. That's in the scientific literature from the 50s.

People like Sidney Siegel, or, you know, my father, Alvin Weinberg, director of Oak Ridge, fired for doubting allow safety, embrittlement in nuclear plants, around 1970.

What is San Onofre? It's a crime scene. What's the crime? Well, if I ripped off one of you for $5.3 billion, I'd be in prison forever, even after I was dead.

So the question that has to be asked is what

happened? The surfers at San Onofre, which I'm too fat and old to be one of, for thirty years wanted the reactors removed.

So what's the rush? And by the way, what happened in San Onofre isn't the worst thing. There's an old paperback about a meltdown, like Fukushima -- that would mean evacuation of Riverside, San Diego, and Orange County for centuries. It would bankrupt the United States if no one is killed, and that's a big "if."

Who did it? Well, I don't know. Certainly Southern California Edison had something to do with it -- they crushed EPRI, EPRI is their public relations arm. I was interviewed by Chauncey Starr (phonetic) and Ed Zebrowski (phonetic), and when they heard about alloy 182 weld embrittlement, they were horrified, I was on an interview.

They didn't hire me. I worked for a nasty Jewish P-R-I-C-K named Hyman Rickober (phonetic) -- he wasn't just afraid of Soviets, he was afraid of everyone, he was paranoid schizophrenic. He knew what he was doing. He won the Cold War from us, as this fellow from the nuclear Navy can attest.

I have two books I'd just like to bring to people's attention, and by the way, if you want to find out about me, Google "A-N-N-A Mayo, If Leaks Could

Kill," and if you want to find an article by me, go to flickr.com and put in the word "giant magneto-resistance read page 312."

This is a book on fractography. What you do is you look at things with a microscope, just like a crime scene where someone is murdered, and you see what you see.

Once it's put away in Idaho, they'll never find it again. It has to be examined here. Why? Because there's $5.3 billion in bills we're going to have to pay. Who is going to pay them?

My feeling about nuclear fuel is that people who produce it should eat it for dinner, and the Securities and Exchange Commission is watching very carefully.

Lastly, in closing, a very interesting book which you all can buy online -- it's called De re metallica, I'll show you the front page, you can buy it from Dover for 30 dollars, by Georgius Agricola, it's the first book on metallurgy.

It talks about why Roman -- you see, there's nothing needed, you don't need anything hard in a nuclear reactor, any woman knows this, your cuticles when you're skiing -- hard things are brittle, they brink.

You're not machining with a nuclear reactor, or nuclear weapons, or jet engines.

De re metallica, translated by Herbert Clark Hoover, President of the United States -- publication date is interesting, 1550, half a millennium ago. Hard things break. Roman swords which stab through Hebrew and bronze shields, and Greek, broke, because they were brittle.

Why doesn't DOE, where the sins come from -- NRC shouldn't be badly blamed -- I'm, my thinking is Soviet sabotage, when I was a little boy.

Not my father or Alvin Weinberg, not Eugene Wigner who founded it -- I have some names, I've been looking at this for 40 years.

How come the NRC, but especially DOE, the only person doing anything in the DOE is Lofaro, look up anything by Robert Lofaro, alloy embrittlement mitigation -- and what happened at San Onofre may have saved us from a much worse thing, core meltdown.

But why should we pay for it? If someone would like to pay my part of the stranded costs, I'll give you my name, if anyone wants to give me a business card, I can email you lots of stuff. Thank you.

FACILITATOR CAMERON: Okay, thank you. Thank you, Ed.

(Applause.)

Yes, sir.

MR. JOHNSON: Thank you. Roger Johnson, San Clemente.

Earlier this evening you showed a slide, I think it was slide 20, on which you listed the reasons why this plan should be denied.

One of the reasons was that it would endanger public health and safety. And I think this plan does endanger public health and safety.

And I think a lot of other people do too, and one of the things that has been ignored in your report, and it's been touched on here briefly, I'd like to focus on this, is terrorism.

The National Academy of Sciences was very concerned about this. In 2006, they wrote a report called Safety and Security of Commercial Spent Nuclear Fuel Storage, and they addressed the whole report on that.

I noticed on slide 38, you ignored terrorism. You listed all the possible things that could go wrong, and you ignored terrorism.

One of the first things that the National Academy of Sciences said is that nearly all the studies on the dangers of nuclear power plants focus on

accidents related to equipment failure, and they completely ignore some of the other ones.

So after 9/11, terrorism became a reality. So here are a few things that they said. They say, attacks, this is page 35, attacks by a knowledgeable terrorist with access to advanced weapons might cause considerable physical damage to spent nuclear plants.

Then they go on to point out the U.S. commercial nuclear power plants are not required by the NRC to defend against air attacks, that's page 31.

They go on to say that nuclear power plants are not designed to resist external terror attacks. There are currently no requirements in place to defend against large-scale terrorist attacks.

Then they go on to say, this committee, the National Academy of Sciences, judges that some attacks involving aircraft would be feasible and could be carried out, and a zirconium cladding fire would melt the fuel pellets, could release some of the radionuclides in the atmosphere, and could be transported hundreds of miles downwind.

Some of the other observations about dry casks. Dry casks were designed to ensure storage, they were not designed to resist terrorist attacks.

And when they talk about fuel pools, they

said, "This Commission concludes that there are scenarios that could lead to partial failure of the spent fuel pool walls, thereby resulting in partial or complete loss of fuel pool coolant."

So it goes on and on, and the one other thing they noticed is that a problem with terrorism is they're site specific, and the NRC has come up with this generic plan, which means that all plants are equal.

So it does not recognize that you have chosen to store waste on earthquake faults in a tsunami zone, in the middle of metropolitan areas, and in an area which is easily accessible by terrorists, it's two or three hundred feet from public highways, any truck bomb could go in there.

The Commission also said that to defend against truck bombs, you needed to have at least 400 feet of setback. Old Pacific Highway is 300 feet, anybody can drive and park there and blow up a truck bomb. There's hardly a day in the week when truck bombs don't go off somewhere in the world.

So this is a very dangerous site that you've chosen, and I think one thing that you should -- should be considered, and is not being considered, is that this is meant to be a temporary long-term, an oxymoron, storage facility, and if you're going to store it

long-term temporary, let's store it long-term temporary somewhere else, like in a remote region where it's safe.

You don't need a permanent solution.

(Applause.)

FACILITATOR CAMERON: Thank you, thank you Roger. We're going to go over to this side of the room, and we'll be coming back to you.

But let's -- let's go to Dan. Dan?

MR. DOMINGUEZ: My name is Daniel Dominguez, I am the Chief Officer for the local union that represents the maintenance and operators at the plant.

I'll keep my statements short. I have worked at the plant for 32 years, 25 years as a reactor operator, and of those 32 years, my primary goal was to operate -- was to protect the health and safety of the public, and in the process, generate electricity for the benefit of society.

There has been a lot of talk. I want to cover one point is that in my 32 years, I have had lots and lots of interaction with the NRC.

And in my interactions, I've always found them very professional, and I've always found them that they had the same goal that I had, which was to protect the health and safety of the public.

When the gentleman, I forget who it was that said they put no price on safety, they do not put a price on safety. I can attest to that. That's been my experience.

And finally, the question about whether it's safer in the pool or safer in the ISFSI. For the last 15 years I've monitored the ISFSI pad, and it is safer in the ISFSI. Thank you.

FACILITATOR CAMERON: Okay, thanks Dan. We're going to go to Patricia and Reuben, and Carlos, we're coming back to you, I didn't forget.

MS. BORCHMANN: Thank you. My name is Patricia Borchmann, I live in Escondido.

I am concerned about the safety of this proposed decommissioning plan that Edison has prepared, and they are trying to expedite, and, you know, make it appear as if it's no, not going to be any problem, or nothing has been overlooked, or nothing has -- there are no unforeseen risks.

I disagree. I think that a lot of the technical comments that have been made are very credible, and I think -- well, I think arguments on both sides have been made that are very technical and very credible, so if I understand there's a dilemma, and, you know, the job is not finished.

But what I would like to emphasize is that you have the authority right now, there's nothing holding you back -- you don't have to wait 90 days to -- until this 90 day limit is up, to ask for additional information. You have that authority right now.

And I think that based on the concerns that have been presented at your Community Engagement Panel through the series of public meetings held by Southern California Edison and their technical experts, there has been plenty of technical, highly technical, incredible concerns raised that aren't covered in Edison's plan.

Edison -- you know, NRC, you are saying, as a lot of people have said, they put no -- there is no cost placed, there is no price limit placed on safety.

I agree with that, because Edison, Edison is supposed to be the one absorbing the cost issues, and they're not. This plan they prepared is the shortcut, walk-away, cheapest possible method. And it's not good enough for southern California.

And there's no reason that -- I don't think people realize, you know, they have better options, that are used internationally, that provide the kind of protections that this densely populated area deserves.

Thank you.

(Applause.)

FACILITATOR CAMERON: Thanks, Patricia. And Reuben?

MR. FRANCO: Thanks, Chip.

Thank you all for letting me speak here today. I am Reuben Franco, I am the President and CEO of the Orange County Hispanic Chamber of Commerce, and I live in south Orange County.

Given that we have a lack of leadership in Washington for any long-term solution to the problem of long-term storage, it's my belief that moving the spent fuel like the plan suggests, from the fuel -- from the pool to the cask storage, would be a lot better solution, a lot safer solution, and hopefully we can move down that road.

So I'd like to thank Edison and the NRC and the employees there for doing their part and trying to come to a solution to this and a resolution, so thank you.

(Applause.)

FACILITATOR CAMERON: Thank you, Reuben.

And let's go to this gentleman right here, and then we'll go over to Carlos, and then I'm going to see if I can find some people.

MR. ALDINGER: Thank you.

Hi, my name is Karl Aldinger. I am a concerned citizen from Fallbrook.

My question is, will there be a contingency plan for when, inevitably, a cask cracks or fractured?

I know Mr. Csontos, you mentioned the ability to weld them. It appears to me that that's the way these are formed in the first place. You create the container, insert the fuel, weld it, so naturally, you should be able to weld up any problems you have.

Has the commission planning process considered having additional storage containers manufactured ahead of time so if we do have a catastrophic failure, you're not standing there looking around for who is going to create this thing that could take five years to create?

We've seen, in the past, it takes an enormous amount of time to build stuff that you guys require.

Our steam generators took five, seven years, and I don't expect these containers to take that long, but who knows what situation we're in 40 years from now. It may not be that easy for us to get more of those, and so it may make sense to try to procure those now.

I wanted to point out that Hanford and WIPP both indicate that a failure to plan for contingency

creates a toxic mess that's very hard, if not impossible, to attend to quickly.

So I hope that, in your thinking, that it's not we build the casks, we build the cement surrounding, and then we pray. I hope that there's an easy way to clean up any problem that you do determine, that there is made with a cask -- that you've thought about this ahead of time and said, it's okay, we've got this. We're not going to send in troops for four or seven months to go slowly clean up a mess.

In talking about the desalination plant, it's worth noting that the Western hemisphere's largest desalination plant is currently being constructed five miles from this room we are standing in.

That plant will provide human drinking water to many sitting in this building. And unlike Japanese officials, who have been very slow at lying through their teeth about the amount of emissions their ongoing nuclear disaster is leaking directly into the Pacific Ocean three and one half years after the earthquake and tsunami, we will not be so polite if and when an accident at San Onofre spent fuel storage poisons our drinking water.

The reverse osmosis system being built there is not designed to and is not likely capable of

filtering out radionuclides. Thank you.

(Applause.)

FACILITATOR CAMERON: Okay, thank you. So do you want to offer anything about what happens if a cask breaks, Al?

MR. CSONTOS: Sure. So first, what we're trying to do in these aging management programs is to set up the guidelines, or the guidance, that says, you know, this much and no further, okay, in terms of degradation and what is acceptable, okay?

That's first. Second is, is that stainless steel repairs have been done in the nuclear industry for decades, probably longer than I've been alive.

And it has -- they've had thousands, in the 80s and 90s, thousands of welds that have had cracking in the reactor system, the stainless steel, same stainless steel that these are made out of and others are made out of, that have been repaired.

Overlays have been done for decades. So there are technologies out there. When there are unique issues, like what happened, and I've been bringing up Koeberg, that's a plant that had, that's right near the coast, breaking waves right next to it, in South Africa, that had this same issue.

And their tank was very thing, so the typical methods of doing repairs was not appropriate, okay?

So they went off and they did some quick R&D and got repairs done very quickly. It was an overlay, okay, but it was a unique overlay.

So the capabilities out there are, they're there. They're there now, okay?

Just bringing it to bear to this side of the house, which is the dry cask storage side, is not that big of a deal, in my opinion.

FACILITATOR CAMERON: Okay.

MR. CSONTOS: But, and then the other issue was, you said, for casks that have been damaged beyond our threshold, or to a point where, you know, we need to do something.

We have casks that are transfer casks, as well as transportation casks, that are at different sites that we could pull together, or, you know, sometimes they are on individual site-specific sites -- site-specific licensees will tend to have them right there on their site, and they can be stored within them and held and stopped, if we're having any issues.

FACILITATOR CAMERON: All right. Thank you, thank you Al. Carlos?

MR. OLVERA: Thank you. I'll try to make this short.

I am referring to off-site emergency planning. My name is Carlos Olvera (phonetic) from the city of Dana Point.

When we became a city in 1989, Southern California Edison installed an emergency response center at city hall. We have not had to use that for San Onofre, but we did use it a couple years ago when we had a tsunami. It only measured six inches, but nevertheless, it was nice to have it.

So I would just ask you, will that facility be maintained throughout decommissioning?

MR. PALMISANO: Let me speak to that.

We've submitted a defueled emergency plan for, you know, that is based on the scenarios that can occur in decommissioning plants.

We will, we have some offsite facilities the utility maintains. The offsite facilities that the counties and cities maintain will really be up to them in the future.

We've talked to the Interjurisdictional Planning Commission, which is the agency which coordinates all offsite emergency planning, not just for nuclear issues, but for all hazards, and that

question really is for them.

They are certainly going to keep their emergency capabilities in place while the fuel pools are in service.

FACILITATOR CAMERON: Okay, and we do have someone from FEMA, but -- the Interjurisdictional Planning Committee, they couldn't be here tonight, but they did give me something that they wanted me to read into the record, very short.

"The members of the SONGS Interjurisdictional Planning Committee have committed to maintaining emergency response capabilities related to nuclear preparedness throughout the SONGS decommissioning process, and to continue our multi-agency partnership to accomplish this goal."

And I'll read more of this if we have time, but I just wanted to get that on the record, and if you could just introduce yourself to us.

MR. GRUNDSTROM: My name is Richard Grundstrom, and I am the Technological Hazards Branch Chief for FEMA Region IX.

And part of what we do is we oversee the radiological emergency preparedness program around the offsite agencies around the nuclear power plants.

The IPC, all the Interjurisdictional

Planning Committee and all the local communities, have a very, very robust emergency response plan, and it's been exercised for years in its -- in a transition over the period of time, like the gentleman mentioned, it was activated for the tsunami. It's now becoming an all-hazards plan.

Just because the San Onofre Nuclear Power Plant is going to shut down, the plan is going to remain in place. They are still going to have the robust EOC (phonetic), and they're still going to have the planning efforts that they do now.

FACILITATOR CAMERON: Okay. And we're going to go to Jacqueline Woo (phonetic), and I'll just read more of the Interjurisdictional Planning Committee as I'm walking over there.

"As a part of our ongoing emergency planning, we will retain the ability to receive information, independently monitor and assess conditions, and take actions to protect our residents, visitors, and emergency workers." This is Jacqueline.

MS. WOO: Thank you.

I have been following the San Onofre whole spectacle through the news, and it's really eye-opening to come here and to hear form all these different perspectives, especially from those who actually work

onsite, who put their lives on the line.

My question is, is there a list of other agencies that we're working in collaboration with for the decommissioning effort, and if so, how can the public participate, will there be any public hearings?

FACILITATOR CAMERON: Okay. Anybody want to take that on?

I think it was a pretty straightforward question. Do you understand it up there?

MR. BROADDUS: When you're talking about other -- other agencies, are you talking about federal agencies, state agencies?

I mean, what -- ?

FACILITATOR CAMERON: I think any, any other agencies that might be involved.

MR. PALMISANO: Let me just make one quick comment.

We've focused our discussions tonight on NRC requirements for decommissioning and the Post-Shutdown Decommissioning Activities Report, the Irradiated Fuel Management Plan.

Realize also, the state of California, through the California Environmental Quality Act, also has some permitting reviews they will do, similar to what we did on Unit 1 activities in some cases.

So there will be some state agencies involved. That is, you know, starting in the near future.

I don't have a list per se. But if you pay attention to our songscommunity.com website as we proceed through the state permitting process for decommissioning activities, that information would be available.

FACILITATOR CAMERON: Yes, and FEMA, FEMA obviously, right. Go ahead, Larry.

MR. CAMPER: No, I was going to say that, for example, Department of Transportation regulations that apply to the waste that will be leaving the site when decommissioning is going on, we actually enforce those regulations, but they are Department of Transportation regulations.

There are also certain EPA considerations that we carry out as part of our regulatory process.

With regards to the hearing question part of it, I mentioned that when the License Termination Plan is submitted, there is an opportunity for a hearing. And if a body, if a group or an individual seeks a hearing and is granted standing, then it goes to an adjudicatory process. But that's an actual legal hearing.

FACILITATOR CAMERON: Okay.

MR. CAMPER: It's not a public exchange of information, it's an adjudicatory hearing.

FACILITATOR CAMERON: As opposed to a meeting. Thank you, Larry.

MR. BROADDUS: Yes, if I could --

FACILITATOR CAMERON: Let's go -- go ahead, Doug.

MR. BROADDUS: -- if I could say one other thing, which is just that for any of the other licensing amendments that San Onofre has submitted to us for review as well, it's just like with the License Termination Plan, there's an opportunity for a hearing provided on each of those as well.

So anyone who is interested or, you know, wants to participate in that can make their request for those as well, such as the emergency plan that Mr. Palmisano talked about previously, that's one of the licensing actions that are under review right now.

FACILITATOR CAMERON: Okay, thank you. Thank you, Doug.

And we're going to go to this gentleman here, but I wondered, is Steve Adams, Francis Bauer (phonetic), or Dave Peiser still here?

Okay, Dave?

MR. PEISER: I am Dave Peiser, and I am running for Congress right here in the 49th District, and San Onofre is obviously part of this District, and I am running against Darrell Issa, in case you wanted to know that.

So the first thing I want to say is thank you for all your hard work to make sure that -- to make sure of the safety and well-being of our District, with all the actions that you are taking, and I want to thank everybody who is in this audience, too, to bring your concerns.

And the one concern of mine is that, according to the timeline, the Department of Energy is going to take this fuel offsite, and I'm concerned that that's never going to happen, considering the history that has been going on so far with trying to find a permanent site for nuclear spent fuel.

So for that reason I have two points. One is I have, because of my serious concerns, I'd really like to see us put a plan in place to get the fuel offsite as soon as possible.

And number two, if you cannot figure out a way to do that, I really think we should look at a longer term containment strategy with the cast iron type casks, something more permanent and durable that's going to

last longer than the timeline that you've put together.

Thanks.

(Applause.)

FACILITATOR CAMERON: Okay, thank you. And let's go to Ted, Ted Quinn.

MR. QUINN: Hi, I am Ted Quinn, and also a CEP member, like Dan, and Gene Stone, I think.

I'd like to thank the NRC for sponsoring this meeting. It's a pleasure to see the factual data, and I'm going to emphasize the word factual data, being presented.

In the CEP we've had multiple meetings, two workshops, to foster public and plant owner exchanges on key issues, and it's continued tonight, even though this isn't a CEP meeting.

The public interactions have been great to see, particularly the ones that are on a factual basis.

I would just like to comment, if Dr. Siegel is still here, there's a good book about threats called One Second After by William Forstchen on the loss of electricity in a North Carolina town, that I think is quite an interesting read.

Thank you.

FACILITATOR CAMERON: Okay. Thank you, Ted.

Francis Bauer? Steve Adams?

Okay, and Alice has left. Do you want to ask that question, Sharon?

Okay, we have a request for clarification from Sharon on a couple of dates. Go ahead.

MS. HOFFMAN: The gentleman who spoke about the DOE and the waste confidence gave a date of 2048 for building an interim facility, and yet Mr. Palmisano talks about all the waste being removed by 2049.

There is something wrong with those dates. And I would just like somebody to clarify how those two dates work together.

FACILITATOR CAMERON: Okay, and I think -- I think Keith's 2048 was based on what DOE actually stated, they thought a repository would be ready in 2048.

So why don't you talk, and then we'll ask --

MR. MCCONNELL: The distinction is that there were two facilities in DOE's strategy.

The first was a centralized interim storage facility. That would be located somewhere in the U.S. And that would occur in the 2020s, so the fuel would be removed from the reactor facilities to this centralized

facility.

That would then be staged for a repository that would be available in 2048 for the final disposition of spent fuel.

FACILITATOR CAMERON: Okay, and Tom, do you want to add anything?

MR. PALMISANO: No, I think that clarifies it.

The dates we're working from are based on Department of Energy information for the interim facility, so for a planning basis, that's what we're using at this point.

We're obviously monitoring the situation. The 2048 is a permanent repository. So our goal, quite frankly, is for the fuel to be removed as soon as DOE can remove it to an interim facility, and that's what our dates are based on.

FACILITATOR CAMERON: Okay. And I just wanted to thank all of you, you've been a tremendous, tremendous group, and I am going to turn this over to the senior NRC official, Larry Camper, to close the meeting out for us.

Al Csontos will be held hostage after the meeting, okay?

So go ahead, Larry.

MR. CAMPER: Thank you, Chip.

Before I make my closing comments and observations, I do want to ask Duane (phonetic), who is in our Office of Nuclear Security Incident Response -- several times terrorism has come up.

And I asked Duane to say something about a concept called safeguards. We have other ways than environmental impact statements or safety reviews for addressing terrorism, and it's under the umbrella of safeguards.

So Duane, would you make a few comments about that to clarify for people how that works, without getting into the details you can't get into?

DUANE: Yes, I'll try to. That's one thing about security, a little bit more sensitive subject, so we don't talk about it as much.

But one thing that's been mentioned a lot during our discussions is that security is going to go away, or go down -- so I do want to let you know that NRC does have a process, and so security will remain in place.

We have a high -- we require high assurance that security -- that all of the sites, from operating, as it starts off operating, to decommissioning through the ISFSI, that they maintain a specific level to go

against what we call our design-basis threat.

And so that design-basis threat is basically scenarios that have been made up of different types of threats, and each reactor, each decommissioning site, has to make sure they maintain those -- their security at that level.

When we say that security is being changed, basically what we're saying is because the operations of the facility from an operating facility to a decommissioning facility changes.

You reduce the area, because, for example, you no longer have the reactor, you no longer have the auxiliary equipment that protects the reactor, so of course you don't have as much area to have to secure, so it's not that security goes away, it's just that it changes the instructions so that we can maintain that same level of security.

So we wanted to make sure it was clear that security is going to be there. The NRC does a very good job, we have a whole office dedicated to ensuring that different plants, including SONGS, will maintain their security even as they make changes -- they have to submit those changes to us, and we do review those changes to make sure that they still have that high level of assurance.

So I just wanted to make that point so that it's clear that the site will still be protected, and that's what, also, Doug was saying in regards to the difference between spent fuel pool and the dry canisters, and that the plant itself will be still secured. So that's why it wouldn't matter.

So I hope that helps.

MR. CAMPER: Great.

FACILITATOR CAMERON: Thank you.

MR. CAMPER: Thank you, thank you.

FACILITATOR CAMERON: And Larry.

MR. CAMPER: Okay, thank you.

First of all, let me thank everyone for all the comments -- very insightful comments, and let me assure you that the meeting is being transcribed. The staff will review the transcript as we go through our review process, you know, tracking against the 90 day clock that's been mentioned several times over the evening.

The staff will caucus following this meeting and discuss the various things that we heard, and we'll caucus as we look at the transcript, and your comments will in fact be considered as we conduct our review.

The second thing I want to mention is, and

I'll go back to the term who can say no? Who can say no? We can say no.

It is correct that we do not approve the PSDAR, consistent with the existing Commission policy. Why is the policy that way? Because in 1996-1997, the Commission determined that the activities that are conducted during decommissioning are bounded by the operations and safety considerations that take place during an operating reactor, there's nothing that is taking place during decommissioning that is extraordinary as compared to the safety and environmental considerations of an operating reactor.

And therefore the Commission put in place the process that we have today, whereby the PSDAR would be submitted, certain information would be provided in that PSDAR, and you've seen the contents of that in some of the slides today -- and then the emphasis is put then upon the ultimate end state of the site.

What does the site look like from a radiological standpoint when that license is prepared to be terminated?

Now, one can criticize that process, I understand that. I am merely offering an explanation as to why it is the way that it is.

Now we ask questions, we have asked

questions, of other PSDARs in the past that have been submitted. We may find ourselves asking questions about this PSDAR.

But I want to point out that the reason we ask the questions, and the reason that we review the PSDAR, is to ensure that our regulations are in fact met.

We have the authority to stop this decommissioning or any other decommissioning at the PSDAR state if we can't get answers to the questions that lead us to believe that our regulations would be complied with. We have regulatory tools that would let us do that.

So we have the authority to say no, even though we don't approve the PSDAR as such, for the reasons I've just explained.

So do understand that we do have that authority to say no.

There's been a great deal of talk about moving the fuel. I think we all think that that's a very legitimate concern. Our country, our -- we do not have a national policy at this point in place that leads to moving of the fuel to another location from the coast here.

I think we all prefer, would prefer, that we did. But that's a national policy decision. It's

not a decision that SCE can make, it's not a decision that we can make, it's a national policy issue.

What we have to do is make sure that this decommissioning takes place in a safe manner, and that the fuel that remains in dry cask storage on the pad is done safely and in a way that will protect public health and safety.

I hear a great deal of interest in the cask of choice, in the cask performance considerations. Those are very fair questions, those are very fair concerns. And we'll do everything that we can to continue to put information on our website that will enunciate the various studies and things that we're working on that Al commented on in his presentation as he answered questions.

Why exemptions? Another fair question. The reason that we grant exemptions for things such as emergency preparedness and certain security considerations, operator qualifications when reactors move into decommissioning, is because our regulations currently in Part 50 are designed around an operating reactor.

We have not yet put in place a set of regulations that would be appropriate once the reactor had gone from operations into decommissioning.

The reason that we haven't is in the year 2000 the staff started down the pathway of a Rule that would articulate specifically what are the conditions that the reactors must be in when in a decommissioning mode. That Rulemaking was put on the back burner, it was postponed, by the Commission, because at the time, it was determined that we had higher priority Rulemakings to work on that dealt with security in a post-9/11 environment.

So, I'll stop there. I think there has been some excellent comments. We thank you for those. We will consider them. And we'll look forward to communicating with you more as we go through the process.

FACILITATOR CAMERON: Okay. And thank you, Larry.

And here's where you can submit email comments or hard comments, hard copy comments, right there. That will remain up there, and there are copies of the Post-Shutdown Report, compact disks out on the table there. So, thank you.

(Whereupon, the meeting went off the record at approximately 9:00 p.m.)

NRC ISFSI Pad Surveys at San Onofre Nuclear Generating Station

Scope:

To perform independent measurements and verifications of radiological conditions at San Onofre Independent Spent Fuel Storage Installation (ISFSI). The measurement locations included background areas, public access areas in the owner controlled areas, protected areas of the facility, and representative measurement areas on both generally licensed ISFSI Pads: Transnuclear, (TN) Inc. Nuclear Horizontal Modular Storage (NUHOMS) and Holtec HI-STORM UMAX dry fuel storage systems.

Equipment Used:

The NRC used a Ludlum Model 19, a sodium iodide instrument calibrated to Cesium-137, which is representative of the gamma energy level at the facility. The instrument used was NRC Tag Number 033906, serial number 84259, with an annual calibration due date of July 23, 2019.

Survey Methodology:

On October 22, 2018, all surveys were performed by the NRC Inspectors.

The survey methodology for the Holtec HI-STORM UMAX systems consisted of a gamma exposure rate at 16 points on each Vertical Ventilated Module (VVM) that was loaded. The points encompassed 8 inlet air vents (Figure 1), 4 points on the closure lid at locations representing 0º, 90º, 180º, and 270º (Figure 2), and at 4 points on the outlet air vent at locations representing 0º, 90º, 180º, and 270º (Figure 2).

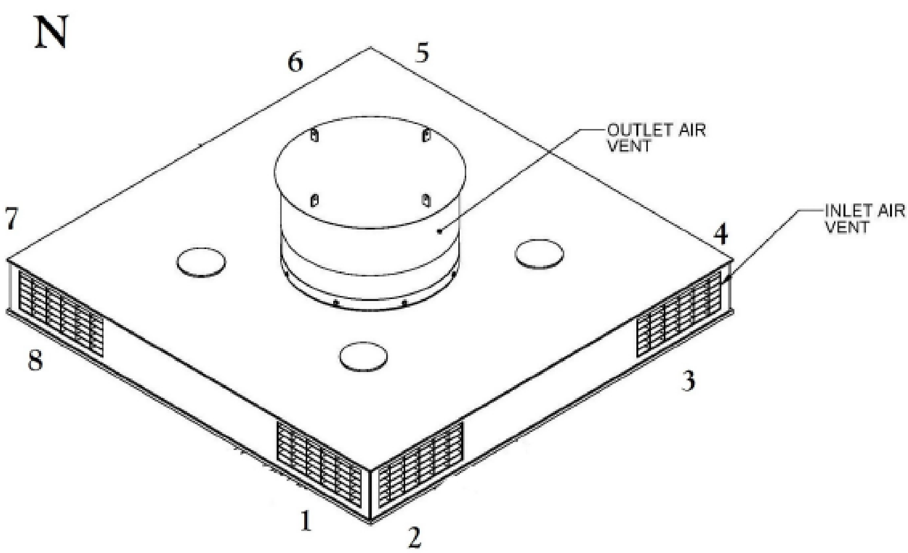

Figure 1

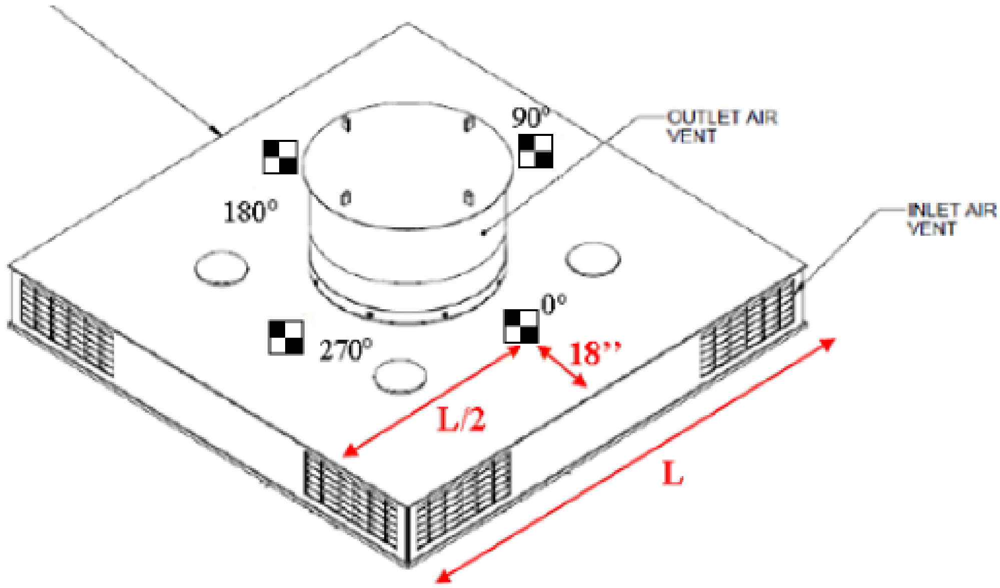

Figure 2

The survey methodology for the TN NUHOMS systems consisted of a gamma exposure rate survey at two points on the air inlet vent for each loaded TN system. These measurement points consisted of a contact survey measurement on the air inlet vent and at 1 foot away from the air inlet vent (Figure 3), for each of the loaded TN systems.

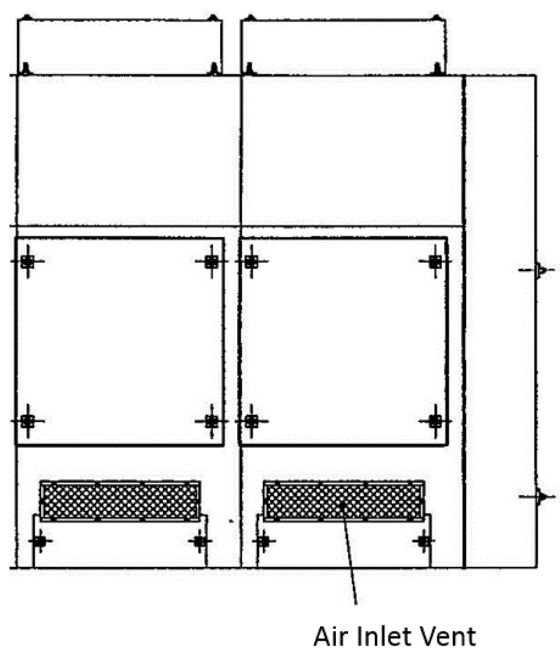

Figure 3

In addition, the NRC performed survey measurements at many locations around the ISFSI pad, Unit 1 Reactor Pressure Vessel, behind the sea wall of the plant, in the upper parking lot of the plant access, and along the bluff overlooking the ISFSI pad. The NRC identified the survey measurement location for each data point on maps of the site layout and documented the data on data point sheets for each of the loaded canister systems.

Results of NRC Surveys:

Holtec HI-STORM UMAX:

Highest gamma measurement was from VVM #33:
Inlet air vents: 330 μR/hr
Closure lid: 18 μR/hr
Outlet air vent: 240 μR/hr

Lowest gamma measurement was from VVM #22:
Inlet air vents: 160 μR/hr
Closure lid: 15 μR/hr
Outlet air vent: 120 μR/hr

General area gamma exposure rate on the Holtec UMAX pad ranged between: 8-15 μR/hr

TN NUHOMS:

Highest gamma measurement was from TN #21:
Inlet Vent on Contact: 1,600 μR/hr, 1 Foot Away from Inlet Vent: 1,100 μR/hr

Lowest gamma measurement was from TN #15:
Inlet Vent on Contact: 100 μR/hr, 1 Foot Away from Inlet Vent: 70 μR/hr

General area gamma exposure rate on the TN pad ranged between: 4-14 μR/hr

Other survey measurement results include the following locations:

Unit 1 Reactor Vessel: Highest Gamma Measurements: 20 μR/hr
Lowest Gamma Measurements: 6 μR/hr (both sides)

Along the beach behind the sea wall ranged between: 5-10 μR/hr

Along the bluff overlooking the ISFSI Pad ranged between: 3-6 μR/hr

Upper Parking Lot ranged between: 4-8 μR/hr

Southern California Edison Dose Rate Limits:

Holtec HI-STORM UMAX: (Technical Specification 5.3.3)

Locations	Neutron	Gamma	Total
Closure Lid Cover Plate	200 µR/hr 0.2 mR/hr	400 µR/hr 0.4 mR/hr	600 µR/hr 0.6 mR/hr
Outlet Vents	1,000 µR/hr 1.0 mR/hr	1,800 µR/hr 1.8 mR/hr	2,800 µR/hr 2.8 mR/hr

TN NUHOMS:

There are no cask related technical specifications.

Conclusion:

Based on the surveys performed on October 22, 2018, the NRC has no safety or regulatory concerns with the loaded spent fuel canisters on the ISFSI Pads at San Onofre. The independent surveys performed by the NRC using NRC survey instruments which are calibrated annually, were all well below the applicable technical specifications values. The highest dose rate values that were measured were the TN NUHOMS System at 1,600 µR/hr on contact and 1,100 µR/hr at 1 foot away from the Inlet Vent.

Docket Nos.: 50-206; 50-361; 50-362; 72-041
License Nos.: DPR-13; NPF-10; NPF-15

CERTIFICATE OF SERVICE

I hereby certify that on July 20, 2020, I electronically filed the foregoing with the Clerk of the Court for the United States Court of Appeals for the Ninth Circuit by using the appellate CM/ECF system.

Participants in the case who are registered CM/ECF users will be served by the appellate CM/ECF system.

Dated: July 20, 2020

Respectfully submitted,
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No. 20-70899

**IN THE UNITED STATES COURT OF APPEALS
FOR THE NINTH CIRCUIT**

IN RE PUBLIC WATCHDOGS,

Petitioner,

v.

UNITED STATES NUCLEAR REGULATORY COMMISSION,

Respondent.

SOUTHERN CALIFORNIA EDISON COMPANY,
Intervenor.

**INTERVENOR SOUTHERN CALIFORNIA EDISON'S
SUPPLEMENTAL EXCERPTS OF RECORD**

VOLUME 8 OF 8

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INTERVENOR SOUTHERN CALIFORNIA EDISON'S SUPPLEMENTAL EXCERPTS OF RECORD: TABLE OF CONTENTS

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NUREG-1927
Revision 1

Standard Review Plan for Renewal of Specific Licenses and Certificates of Compliance for Dry Storage of Spent Nuclear Fuel

Final Report

Office of Nuclear Material Safety and Safeguards

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U.S.NRC
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NUREG-1927
Revision 1

Standard Review Plan for Renewal of Specific Licenses and Certificates of Compliance for Dry Storage of Spent Nuclear Fuel

Final Report

Manuscript Completed: May 2016
Date Published: June 2016

Office of Nuclear Material Safety and Safeguards

ABSTRACT

This Standard Review Plan is intended for use by the U.S. Nuclear Regulatory Commission (NRC) reviewer. It provides guidance for the safety review of renewal applications for specific licenses of independent spent fuel storage installations and certificates of compliance (CoCs) of dry storage systems, as codified in Title 10 of the *Code of Federal Regulations* (10 CFR) Part 72, "Licensing Requirements for the Independent Storage of Spent Nuclear Fuel, High-Level Radioactive Waste, and Reactor-Related Greater Than Class C Waste."

This Standard Review Plan provides guidance for the review of general information, scoping evaluation information, and aging management information, included in a renewal application. The guidance provides information on review of time-limited aging analyses and aging management programs (AMPs), including learning AMPs that consider and respond to operating experience. The guidance provides example AMPs for welded stainless steel canisters, reinforced concrete structures, and a high burnup fuel monitoring and assessment program. It also provides guidance on considerations for CoC renewals and the general license framework, including guidance on general licensees' implementation of AMPs.

The NRC expects to periodically revise and update this Standard Review Plan to clarify the content, correct errors, and include new information, knowledge, and experience regarding aging management considerations. Comments, suggestions for improvement, and notices of errors or omissions should be sent to the Director, Division of Spent Fuel Management, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001.

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ABBREVIATIONS

ACI	American Concrete Institute
AMA	aging management activity
AMP	aging management program
AMR	aging management review
ANSI	American National Standard Institute
ASME	American Society of Mechanical Engineers
°C	degrees Celsius
CAP	Corrective Action Program
CFR	*Code of Federal Regulations*
CISCC	chloride-induced stress corrosion cracking
CLB	current licensing basis
CoC	certificate of compliance
CRIEPI	Central Research Institute of Electric Power Industry
DOE	U.S. Department of Energy
DSS	dry storage system
EPRI	Electric Power Research Institute
°F	degrees Fahrenheit
FSAR	final safety analysis report
GTCC	greater-than-Class-C
GWd/MTU	gigawatt days per metric ton uranium
HBU	high burnup
HDRP	HBU Dry Storage Cask Research and Development Project
ICSF	interim consolidated storage facility
ISFSI	independent spent fuel storage installation
ISG	Interim Staff Guidance
ITS	important to safety
kJ	kilojoule
LWR	light water reactor
m	meters
mol	mole
NDE	nondestructive examination
NEI	Nuclear Energy Institute
NMSS	Office of Nuclear Material Safety and Safeguards
NRC	U.S. Nuclear Regulatory Commission
OMB	Office of Management and Budget

PM	NRC Project Manager
ppm	parts per million
QA	quality assurance
s	second
SAR	safety analysis report
SER	safety evaluation report
SRP	Standard Review Plan
SSC	structure, system, and component
TEPCO	Tokyo Electric Power Company
TLAA	time-limited aging analysis
TS	technical specification
UFSAR	updated final safety analysis report

INTRODUCTION

This Standard Review Plan (SRP) is intended to provide guidance to the U.S. Nuclear Regulatory Commission (NRC) staff for the safety review of renewal applications for specific licenses of independent spent fuel storage installations (ISFSIs) and certificates of compliance (CoCs) of dry storage systems (DSSs), as codified in Title 10 of the *Code of Federal Regulations* (10 CFR) Part 72, "Licensing Requirements for the Independent Storage of Spent Nuclear Fuel, High-Level Radioactive Waste, and Reactor-Related Greater Than Class C Waste."

To renew a specific license, an applicant (i.e., the licensee) must submit a license renewal application at least 2 years before the expiration of the license, in accordance with the requirements of 10 CFR 72.42(b). To renew a CoC, an applicant (i.e., CoC holder, user, or user's representative) must submit a renewal application at least 30 days before the expiration of the associated CoC in accordance with the requirements of 10 CFR 72.240(b). The NRC may renew a specific license or a CoC for a term not to exceed 40 years, in accordance with 10 CFR 72.42(a), or 10 CFR 72.240(a), respectively.

The NRC-approved DSSs listed in 10 CFR 72.214, "List of Approved Spent Fuel Storage Casks," may be used by any 10 CFR Part 72 general licensee in accordance with 10 CFR 72.212, "Conditions of General License Issued Under § 72.210." The term of a general license is tied to the term of the CoC being used. Within the general license term, each DSS has its own storage term that begins when that DSS is placed into service at the ISFSI (see Appendix F for a discussion of storage terms). When a CoC is renewed, the associated users' general licenses are also renewed. If the CoC holder chooses not to apply for the renewal of a particular CoC or is no longer in business, a licensee, licensee's representative, or another certificate holder may apply for renewal of the CoC in place of the CoC holder.

Both the specific-license and the CoC renewal applications must contain requirements and operating conditions (fuel storage, surveillance and maintenance, and other requirements) for the ISFSI or DSS that address aging mechanisms and aging effects that could affect structures, systems, and components relied upon for the safe storage of spent fuel. Renewal applications must include (1) time-limited aging analyses, if applicable, that demonstrate that structures, systems, and components important to safety will continue to perform their intended function for the requested period of extended operation, and (2) aging management programs for management of issues associated with aging that could adversely affect structures, systems, and components important to safety. Licensees and applicants are encouraged to meet with the NRC staff at public pre-application meetings to discuss their proposed plans for the renewal application.

The technical review of the renewal application is primarily a materials engineering effort. The materials discipline should coordinate its review of the renewal application with other disciplines, such as the structural, radiation protection, thermal, criticality, and quality assurance disciplines, as appropriate, to help ensure that relevant aspects of the application and review have been addressed.

This SRP defines an acceptable method for the NRC staff to review and determine if the applicant demonstrates that the specific-licensed ISFSI or the certified DSS will continue to meet the applicable regulatory requirements of 10 CFR Part 72 during the period of extended operation. The reviewer should be aware that additional interim staff guidance may have been issued to clarify or address issues following the publication of this guidance. This SRP defines

an acceptable method for satisfying the applicable regulatory requirements; it is not a regulatory requirement. An applicant may propose alternate means for satisfying the appropriate regulatory requirements. However, deviation from this guidance in whole or in part may result in an extended NRC staff review schedule.

The NRC expects to periodically revise and update this SRP to clarify the content, correct errors, and include new information, knowledge, and experience regarding aging management considerations. Comments, suggestions for improvement, and notices of errors or omissions will be considered by, and should be sent to the Director, Division of Spent Fuel Management, Office of Nuclear Material Safety and Safeguards, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001.

The guidance document is not intended to be used for the review of other 10 CFR Part 72 renewal applications, such as those for wet storage facilities or monitored retrievable storage installations.

This guidance document is also not intended to be used as the sole guidance for the review of a specific license application for an “interim consolidated storage facility” (ICSF), where a DSS structure, system, and component (SSC) has been in storage at one location for some period of time and then is transported to a second location (ICSF) for subsequent storage. Guidance for review of an application for a 10 CFR Part 72 specific license is located in NUREG-1567, “Standard Review Plan for Spent Fuel Dry Storage Facilities.” However, as the DSS SSCs that will be in storage at a potential ICSF enter the period of extended operation, this guidance document is applicable to the aging management of such DSS SSCs.

Figure A is a flowchart of the specific-license and CoC renewal process.

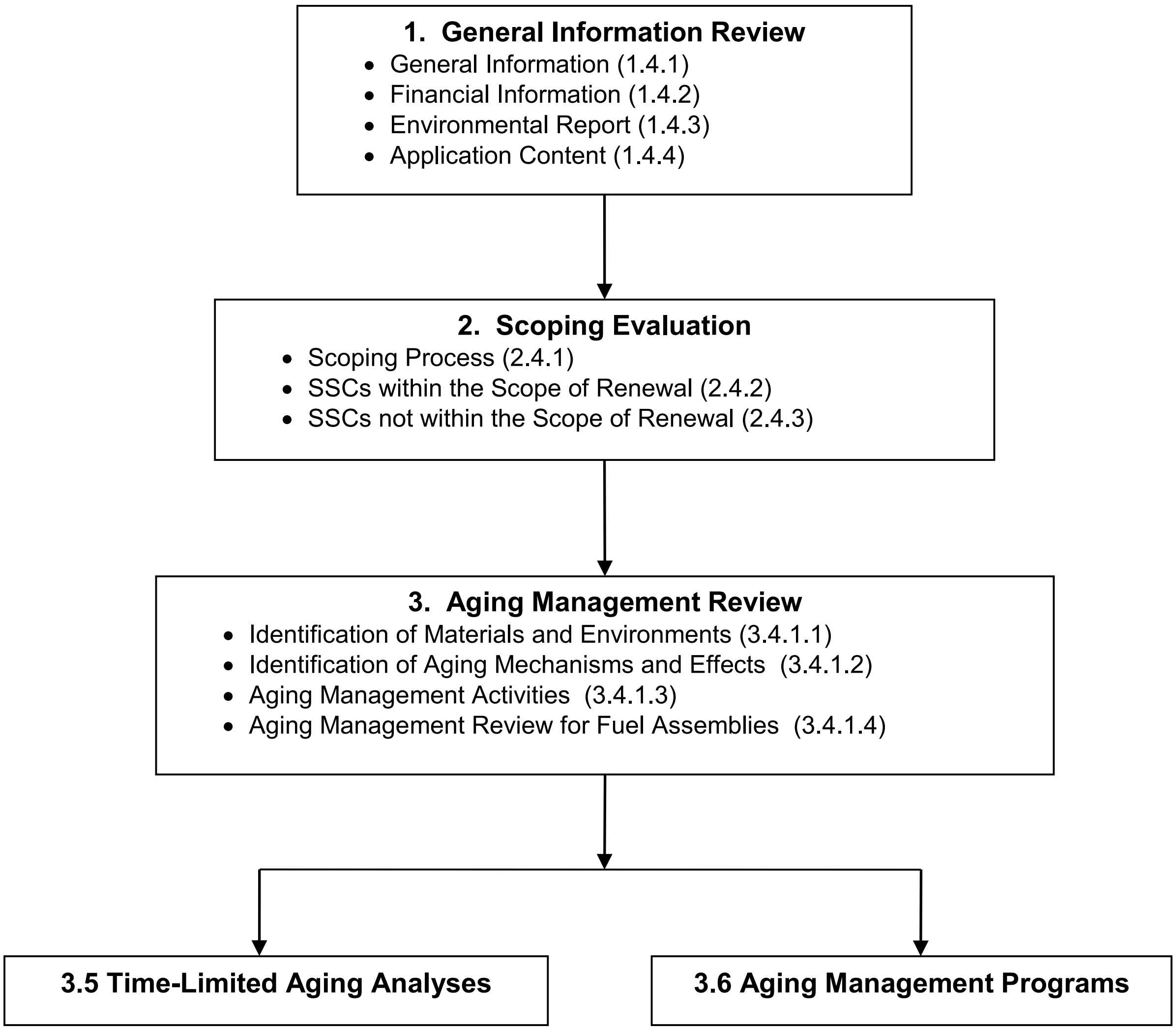

Figure A. Specific-license and CoC renewal process

Revision 1

Based on lessons learned from reviews of specific-license and CoC renewal applications and input received from the public and industry, the NRC staff proposed changes to NUREG-1927, Revision 0, to add greater detail and clarity. The staff held public meetings, including a public meeting on July 14–15, 2014, to solicit stakeholder input on the staff's considerations for revisions to the guidance. The staff subsequently took stakeholder input into consideration and developed the draft NUREG-1927, Revision 1, which was published for public comment on July 7, 2015 (80 FR 38780). The staff considered public comments received on the draft guidance in preparing the final NUREG-1927, Revision 1. The public comments are located in the Agencywide Documents Access and Management System (ADAMS) Package Accession No. ML15356A560. The staff also prepared responses to the public comments, at ADAMS Accession No. ML16125A534.

This revision of NUREG-1927 focuses on expanding guidance on application content, scoping evaluation, aging management review, time-limited aging analyses, and elements of an aging management program (AMP), including evaluation of AMPs and ensuring these programs respond to operating experience to remain adequate throughout the period of extended operation (i.e., learning AMPs). This revision of NUREG-1927 also includes new guidance in the areas of: (1) timely renewal, (2) amendment applications submitted during renewal reviews and after the renewal is issued, (3) use of terms, conditions, and specifications for ensuring AMPs remain adequate during the period of extended operation, (4) commencement of AMPs for CoC renewals and implementation of AMPs, (5) example AMPs, (6) use of a demonstration program as a surveillance tool for high burnup fuel performance, and (7) storage terms (and calculation of length of time that a dry storage system can remain loaded).

This revision to NUREG-1927 was developed in parallel with an ongoing effort by the Nuclear Energy Institute (NEI) to develop guidance for the industry in the preparation of applications for renewal of specific licenses and CoCs. NEI 14-03, Revision 1, "Format, Content and Implementation Guidance for Dry Cask Storage Operations-Based Aging Management" (ADAMS Accession No. ML15272A329), includes guidance on the continued evaluation of operating experience (see Section 3.6.1.10 of NUREG-1927, Revision 1). One of the principles introduced in NEI 14-03 is the use of "tollgates" as a structured approach for assessing operating experience and data from applicable research and industry initiatives. In addition, NEI 14-03 describes an initiative to aggregate and disseminate aging-related operating experience, research results, monitoring feedback, and inspection data between licensees. The staff provided comments on NEI 14-03, Revision 0, "Guidance for Operations-Based Aging Management for Dry Cask Storage" (ADAMS Accession No. ML14266A225), to NEI on January 21, 2015 (ADAMS Accession No. ML15013A201). At the time of publication, the staff was continuing its review of NEI 14-03, Revision 1, for proposed NRC endorsement. However, until a time when NEI 14-03 may be endorsed by NRC, Section 3.6.1.10 of NUREG-1927, Revision 1, provides guidance to reviewers regarding information in NEI 14-03 that may be used or referenced by applicants for specific license or CoC renewals.

Standard Review Plan Structure

Each chapter of this SRP contains the following sections:

Review Objective: This section provides the purpose and scope of the review and establishes the major review objectives for the chapter. It also discusses the information needed, or coordination expected, from other NRC staff to complete the technical review.

Areas of Review: This section describes the structures, systems, and components; analyses, data, or other information; and their sequence in the discussion of acceptance criteria.

Regulatory Requirements: This section summarizes the regulatory requirements in 10 CFR Part 72 pertaining to the scoping process, aging management review, and aging management activities that include the time-limited aging analyses review. This list is not all-inclusive, and the reviewer should be aware that other parts of the regulations, such as 10 CFR Part 20, “Standards for Protection against Radiation,” are assumed to apply to all licensees. The reviewer should read the complete language of the current version of 10 CFR Part 72 to determine the proper set of regulations for the section being reviewed.

Review Guidance: This section discusses the specific technical information that should be included in the application and reviewed for regulatory compliance. The review guidance can be supplemented by interim staff guidance, NUREGs, etc.

Evaluation Findings: This section provides sample summary statements for evaluation findings to be incorporated into the safety evaluation report (SER) for each area of review. The reviewer prepares the evaluation findings based on the satisfaction of the regulatory requirements. The NRC publishes the findings in the SER.

1. GENERAL INFORMATION REVIEW

1.1 Review Objective

The purpose of the general information review is to ensure that the specific-license or certificate of compliance (CoC) renewal application meets the requirements listed in Section 1.3 below.

1.2 Areas of Review

Areas of review addressed in this chapter include the following:

- general information (specific license only)
- financial information (specific license only)
- environmental report (specific license only)
- application content

Areas specifically excluded from the renewal review include the following:

- structures, systems, and components (SSCs) associated with physical protection of the independent spent fuel storage installation (ISFSI) or dry storage system (DSS), under Title 10 of the *Code of Federal Regulations* (10 CFR) Part 72, "Licensing Requirements for the Independent Storage of Spent Nuclear Fuel and High-Level Radioactive Waste, and Reactor-Related Greater Than Class C Waste," Subpart H, "Physical Protection"

- SSCs associated with the ISFSI emergency plan, under 10 CFR 72.32, "Emergency Plan"

1.3 Regulatory Requirements

Table 1-1 presents a matrix that identifies the specific regulatory requirements pertaining to application content, general information about the specific licensee or CoC holder, financial information, and the environmental report. Additional regulatory requirements for the environmental report can be found in 10 CFR Part 51, "Environmental Protection Regulations for Domestic Licensing and Related Regulatory Functions."

Table 1-1. Relationship of Regulations and General Information Review

Areas of Review	72.22 [a] (a), (b), (c), (d)	72.22 [a] (e)	72.30(c) [a]
Application Content	•		
General Information	•		
Financial Information		•	•
Environmental Report			

Areas of Review	72.34 [a]	72.42 [a]	72.48(d)	72.240 [b] (b), (c), (d)
Application Content		●	●	●
General Information				
Financial Information				
Environmental Report	●			

[a] These regulations apply only to specific license renewals per 10 CFR 72.13, "Applicability."
[b] These regulations apply only to CoC renewals per 10 CFR 72.13.

1.4 Review Guidance

This section provides review guidance for general information (Section 1.4.1), financial information (Section 1.4.2), environmental report (Section 1.4.3), and application content (Section 1.4.4). This section also provides information on timely renewal (Section 1.4.5), amendment applications submitted during the renewal review or after the renewal is issued (Section 1.4.6) and license and CoC conditions (Section 1.4.7).

1.4.1 General Information

The U.S. Nuclear Regulatory Commission (NRC) project manager (PM) should ensure that the specific licensee has provided information under 10 CFR 72.22(a)–(d), including the specific licensee's full name, address, and description of the business or occupation. If the specific licensee is a partnership, the application should identify the name, citizenship, and address of each partner, and the principal location where the partnership does business. If the specific licensee is a corporation or an unincorporated association, the application should specify the State in which it is incorporated or organized and the principal location at which it does business, along with the names, addresses, and citizenships of its directors and principal officers. If the specific licensee is acting as an agent or representative of another person in filing the application, the application should provide the above information for the principal. If the specific licensee is the U.S. Department of Energy, then the application should specify the organization responsible for the construction and operation of the ISFSI and describe any delegations of authority and assignments of responsibilities.

1.4.2 Financial Information

In general, the PM should ensure that the renewal application for a specific license contains the necessary documentation regarding financial data, under 10 CFR 72.22(e), which shows that the specific licensee can carry out the proposed activities for the requested duration. Information should state where the activity will be performed, the general plan for carrying out the activity, and the period of time for which the specific license is requested. The PM should

ensure that the renewal application is based only on the approved design bases and does not include additional construction costs beyond the design bases. The application should identify other costs related to activities associated with managing aging mechanisms and effects, and it should identify ISFSI operating and decommissioning costs that have been revised from those specified in the original specific-license application for construction, operation, and decommissioning. In addition, the application should include a decommissioning funding plan that identifies any changes in decommissioning costs and the extent of contamination, pursuant to 10 CFR 72.30(c).

The scope of this standard review plan (SRP) does not include specific guidance for reviewing financial information. Financial reviews should be coordinated with financial reviewers in the Performance Assessment Branch of the Office of Nuclear Material Safety and Safeguards (NMSS) or the Office of Nuclear Reactor Regulation.

1.4.3 Environmental Report

The PM should ensure that the specific-license renewal application contains an environmental report or supplement, as required by 10 CFR 51.60, "Environmental Report—Materials Licenses" and 10 CFR 72.34, "Environmental Report." The supplemental report may be limited to incorporating by reference, updating, or supplementing the information previously submitted to reflect any significant environmental changes, including those that may result from operating experience as related to environmental conditions, or a change in operations or proposed decommissioning activities.

The environmental report should also meet the general requirements of 10 CFR 51.45, "Environmental Report," as applicable. As required by 10 CFR 51.45(c), the environmental report should contain sufficient data to aid the NRC in its development of an independent analysis.

The review of the environmental report should be coordinated with the Environmental Review Branch of NMSS.

1.4.4 Application Content

The reviewer should look for a map or guide to the renewal application to assist in its review, because the format may vary from that of a standard safety analysis report (SAR). The PM or reviewer should verify that the renewal application for both CoC and specific-license renewals contain all of the following sections:

- general Information (specific license only)
- scoping evaluation
- aging management review (AMR)
- time-limited aging analyses (TLAAs)
- aging management programs (AMPs)
- information pertaining to granted exemptions and their implication to aging management

- changes or additions to technical specifications or to the specific license or CoC

- supplement to the final safety analysis report (FSAR), including:

  - scoping results
  - table of AMR results
  - summary of TLAAs and TLAAs' conclusions
  - summary of AMPs

- annotations to show 10 CFR 72.48 ("Changes, Tests, and Experiments") changes since last biannual update as required by 10 CFR 72.48(d)(2) or any other changes from either the SAR pages included with the last approved amendment (or initial) application, or the FSAR, whichever is most recent

For CoC renewal applications that involve multiple amendments, the PM or reviewer should verify that the renewal application also includes:

- a description of the organization of the renewal application as it relates to the different amendments (i.e., for the CoC as a whole)

  - This could be in the form of a guide to the renewal application to identify the sections or appendices applicable to each CoC amendment.

- a clear description of each amendment

  - That is, what each amendment changed from (or added to) the initial certificate (i.e., "amendment 0"), or what each amendment changed from (or added to) the previous amendments, should be described.

- a clear description of the scope and content of the renewal application as it applies to each amendment

  - If there are different SSCs, materials, contents specifications, or environments described in the different CoC amendments, the application should specify any differences in the scoping evaluation, aging management review, TLAAs, and AMPs, for each individual amendment.

A CoC renewal includes the initial certificate ("amendment 0") and all subsequent amendments. The subsequent amendments have the same termination date as the initial certificate. The CoC holder has the option to request that only certain (i.e., not all) amendments under a CoC be renewed. If amendments are not renewed, upon expiration, casks loaded under that amendment would need to be removed from service when they reach the end of their storage term (see Appendix F for calculation of storage terms). As a means to extend the storage term, a general licensee (cask user) also may have the option to apply changes authorized by another amendment to the CoC that has been renewed, to a cask loaded under an amendment that has not been renewed following the requirements in 10 CFR 72.212.

Drawings provided as part of the renewal application should be clear and legible. If information in drawings is unclear or illegible, the PM should ask the applicant for additional, larger or full-size drawings. The reviewer should ensure that dimensions, materials, and other details on

the drawings are consistent with those described in both the text of the renewal application and the FSAR supplement.

The reviewer should verify that the applicant has updated the appropriate drawings to reflect any changes made to the design of the SSCs through the application of 10 CFR 72.48. Reviewers should be familiar with NUREG/CR-5502, "Engineering Drawings for 10 CFR Part 71 Package Approvals," issued May 1998. Although NUREG/CR-5502 was developed for transportation packages, the criteria for drawings are consistent for storage designs and therefore useful to the review process.

If the applicant provided drawings and descriptions as proprietary information in the application and requested them to be withheld from the public, the PM should review the request for withholding and ensure all the necessary information is available for the NRC to make a decision on the withholding request, in accordance with the requirements of 10 CFR 2.390, "Public Inspections, Exemptions, Requests for Withholding." The applicant should also submit a nonproprietary version of the document to be made available to the public.

The reviewer should ensure the specific-license or CoC renewal application does not include any changes to the design bases. Changes to the design bases must be requested through a separate amendment process. However, the renewal application may include editorial changes or corrections that do not change the design bases.

1.4.5 Timely Renewal

To renew a specific license, an applicant must submit a renewal application at least 2 years before the expiration of the license in accordance with the requirements of 10 CFR 72.42(b). To renew a CoC, an applicant must submit a renewal application at least 30 days before the expiration of the CoC in accordance with the requirements of 10 CFR 72.240(b). When the applicant has submitted a timely application for renewal, the existing specific license or CoC will not expire until a final decision concerning the application for renewal has been made by the Commission. Therefore, any DSSs loaded during the initial license or CoC period may remain in service until the review of the renewal application is complete.

1.4.6 Amendment Applications Submitted during the Renewal Review or after the Renewal Is Issued

By regulation, applicants must demonstrate that SSCs important to safety will continue to perform their intended function(s) for the requested period of extended operation as a part of the renewal request. For *concurrent amendment and renewal applications*, the amendment application should include a scoping evaluation and an AMR for that amendment to document the evaluation of the amendment's SSCs (and associated subcomponents) for extended operation, or the renewal application should be supplemented to address the proposed amendment to document the evaluation of the amendment's SSCs (and associated subcomponents) for extended operation. Any *amendment application submitted after the renewal has been issued* (*post-renewal amendment applications)* should include a scoping evaluation and an AMR for that amendment.

For post-renewal amendment applications or concurrent amendment applications that include a scoping evaluation and an AMR, the amendment application should either: (1) show that the in-scope SSCs (and associated subcomponents) described in the amendment are already encompassed in the TLAAs or AMPs included in the specific-license or CoC renewal

application, or (2) include revised or new TLAAs or AMPs to address aging effects of any new in-scope SSCs (and associated subcomponents) proposed in the amendment application. The PM and technical reviewers should verify that the following information is included in the amendment application (see also Section 1.4.4):

- a scoping evaluation that identifies any new SSCs (and associated subcomponents) included in the amendment request and discusses whether the SSCs (and associated subcomponents) are included or excluded from the scope of renewal, following the guidance in Chapter 2

- an aging management review that identifies any applicable aging mechanisms and effects for the new SSCs (and associated subcomponents) within the scope of renewal

- changes to the FSAR, which should include:

  - scoping results and identification of any new in-scope SSCs

  - revised table of AMR results

  - identification of the approved TLAAs (or the TLAAs included in the renewal application, for concurrent amendments) that address the new in-scope SSCs, or identification and a summary of any revised or new TLAAs and the TLAAs' conclusions that support the amendment

  - identification of the approved AMPs (or the AMPs included in the renewal application, for concurrent amendments) that encompass the new in-scope SSCs (and associated subcomponents), or a summary of proposed changes to approved AMPs (or the AMPs in the renewal application, for concurrent amendments) or new AMPs that will apply to the new in-scope SSCs (and associated subcomponents)

For concurrent amendment and renewal applications, if there are different PMs assigned to the renewal review and the amendment review, the PMs and technical reviewers should coordinate across the reviews to ensure that renewal aspects are covered for the amendment. Note that, before proceeding with the review of an amendment submitted *during* the renewal review, the PMs should consider how each review may affect the other, and decide, in conjunction with Branch and Division management, whether to proceed with both reviews, or to delay one review until the other is complete. For additional guidance, refer to Regulatory Issue Summary (RIS) 2004-20, "Lessons Learned from Review of 10 CFR Parts 71 and 72 Applications, (NRC 2004)."

The NRC staff may include a condition in the renewed license or CoC noting all future amendments would need to address aging management.

1.4.7 Terms, Conditions, and Specifications for Specific Licenses and CoCs in the Period of Extended Operation

In renewing a license or CoC, the staff should consider whether any terms, conditions, or specifications are needed to ensure the safe operation of the ISFSI or DSS during the period of extended operation, including but not limited to, terms, conditions, and specifications that will require implementation of any AMPs. Several conditions are likely to be included in the license or CoC.

Generally, NRC staff will renew the license or CoC with a condition requiring the specific licensee or CoC holder, respectively, to incorporate a renewal supplement into the FSAR, as submitted in the renewal application and revised through the review process (see Section 1.4.4). The specific licensee will be required to continue to update the FSAR under the requirements in 10 CFR 72.70. The CoC holder will be required to continue to update the FSAR under the requirements in 10 CFR 72.248.

NRC staff may renew a license or CoC with a condition requiring licensees to implement the activities in the AMPs (e.g., update, revise, or create programs or procedures for AMP implementation) by a specific date (see Section 3.6.3). These programs and procedures will be subject to NRC inspection to ensure they are maintained, implemented, and periodically updated to respond to operating experience, while providing reasonable assurance that the pertinent SSCs will continue to perform their intended functions in the period of extended operation.

As the entirety of the AMP may not be included in the license, CoC, or technical specifications, site procedures for AMP implementation may later be changed without prior NRC review and approval. Therefore, the staff should consider whether additional conditions or specifications are needed in the specific license or CoC to ensure elements of the AMPs, which are the basis of the staff's findings and the decision to issue the renewal, are effectively retained or implemented (e.g., timeframes for development of inspection or examination methods). These conditions should be specific to information in the AMP described in the renewal application which staff relied upon to make the requisite safety findings of reasonable assurance of adequate protection of public health and safety, and the environment.

The entire AMPs may be included as an appendix to the staff's safety evaluation report (SER), for complete documentation of the staff's safety evaluation, findings, and decision to issue the renewal. The staff should include a separate section in the SER that includes any new or modified terms, conditions, or specifications that were added to the license or CoC as a result of the renewal review. This section should include the basis for each new or revised term, condition, or specification, or it should include a reference to the other sections of the SER that discuss the staff's evaluation and findings that form the basis for the new or revised term, condition, or specification.

For CoC renewals, additional terms, conditions, or specifications may be needed to ensure the safe operation of the cask during the renewal term, including but not limited to, terms, conditions, and specifications that will require the implementation of an AMP, in accordance with 10 CFR 72.240(e). Such conditions may include requirements for a future or existing general licensee (cask user) to include in its 10 CFR 72.212(b)(5) evaluation (including the results of the review and determination per 10 CFR 72.212(b)(6) and 10 CFR 72.212(b)(8)) how it will meet the new CoC terms, conditions, or specifications for aging management (see Appendix E).

The NRC staff may also include a condition in the renewed license or CoC to ensure that all future amendments address the renewed design bases for the ISFSI or CoC, including any aging management considerations. See Section 1.4.6 for guidance on post-renewal amendments.

In addition, the applicant may also propose additional license or CoC conditions as part of its application, if these support the technical basis for the application and the proposed TLAAs or AMPs that ensure the safe operation of the ISFSI or DSS during the period of extended operation.

1.5 Evaluation Findings

The reviewer prepares a summary statement and evaluation findings based on compliance with the regulatory requirements in Section 1.3. The summary statement and evaluation findings should be similar in wording to the following example (the finding number is for convenience in cross-referencing within the SRP and SER):

The NRC staff has reviewed the general information provided in the renewal application and supplemental documentation. The NRC staff performed its review following the guidance provided in NUREG-1927, Revision 1, "Standard Review Plan for Renewal of Specific Licenses and Certificates of Compliance for Dry Storage of Spent Nuclear Fuel," and relevant interim staff guidance. Based on its review, the NRC staff finds:

F1.1 The information presented in the renewal application satisfies the requirements of 10 CFR 72.2, 72.22, 72.30, 72.34, 72.42, 72.48, and 72.240, as applicable.

F1.2 The applicant has provided a tabulation of all supporting information and docketed material incorporated by reference, in compliance with 10 CFR 72.42 or 72.240, as applicable.

2. SCOPING EVALUATION

2.1 Review Objective

The scoping evaluation should identify the structures, systems, and components (SSCs) of the independent spent fuel storage installation (ISFSI) or dry storage system (DSS) that should be reviewed for aging mechanisms and effects.

2.2 Areas of Review

The reviewer should ensure that the applicant has included information about the following areas of review:

- scoping process
- SSCs within the scope of specific-license or certificate of compliance (CoC) renewal
- SSCs not within the scope of specific-license or CoC renewal

2.3 Regulatory Requirements

The U.S. Nuclear Regulatory Commission (NRC) bases a specific-license or CoC renewal on the continuation of the approved design bases throughout the period of extended operation. The entire design bases of the specific license or CoC is considered to be renewed when NRC issues a renewed license or CoC. However, the NRC's renewal review is focused on the maintenance of the intended functions of (a) SSCs important to safety and (b) SSCs failure of which may affect a safety function, as discussed in Section 2.4.2. This guidance document refers to such SSCs as those that are within the scope of renewal, and these are the SSCs that are further reviewed for aging mechanisms and effects, as discussed in Chapter 3, "Aging Management Review."

If new safety-related deficiencies in the design bases are discovered, they must be addressed and rectified through the specific-license or CoC amendment process. The renewal process cannot be used to facilitate approval of design changes.

Table 2-1 presents a matrix of regulatory requirements for renewal related to the scoping review.

Table 2-1. Relationship of Regulations and Scoping Review

Areas of Review	10 CFR Part 72 Regulations					
	72.3	72.24 [a] (b), (c), (d)	72.24 [a] (g)	72.42 [a] (b)	Subpart F: Applicable Sections 72.120, 72.122, 72.124, 72.126, 72.128	72.236 [b] Applicable Sections
Scoping Process			●	●		●
SSCs within the Scope of Specific-License or CoC Renewal	●	●	●		●	●
SSCs not within the Scope of Specific-License or CoC Renewal		●	●		●	●

[a] These regulations apply only to specific-license renewals per Title 10 of the *Code of Federal Regulations* (10 CFR) 72.13.
[b] These regulations apply only to CoC renewals per 10 CFR 72.13.

2.4 Review Guidance

This section provides review guidance for the scoping evaluation. Section 2.4.1 explains the scoping process, Section 2.4.2 discusses SSCs within the scope of renewal, while Section 2.4.3 provides guidance for SSCs not within the scope of renewal.

2.4.1 Scoping Process

Figure 2-1 provides a flowchart of the scoping evaluation process. The reviewer should ensure that the application provides documentation of the scoping process that includes the following:

- a description of the scoping process and method for the inclusion or exclusion of SSCs (and associated subcomponents) from the renewal scope
- a list of the SSCs (and associated subcomponents) that are identified as within the scope of renewal, their intended function, and safety classification or basis for inclusion in the renewal scope
- a list of the SSCs (and associated subcomponents) that are identified as *not* within the scope of renewal and basis for exclusion
- a list of the sources of information used
- any discussion or drawings needed to clarify the process, SSC intended functions and safety classifications

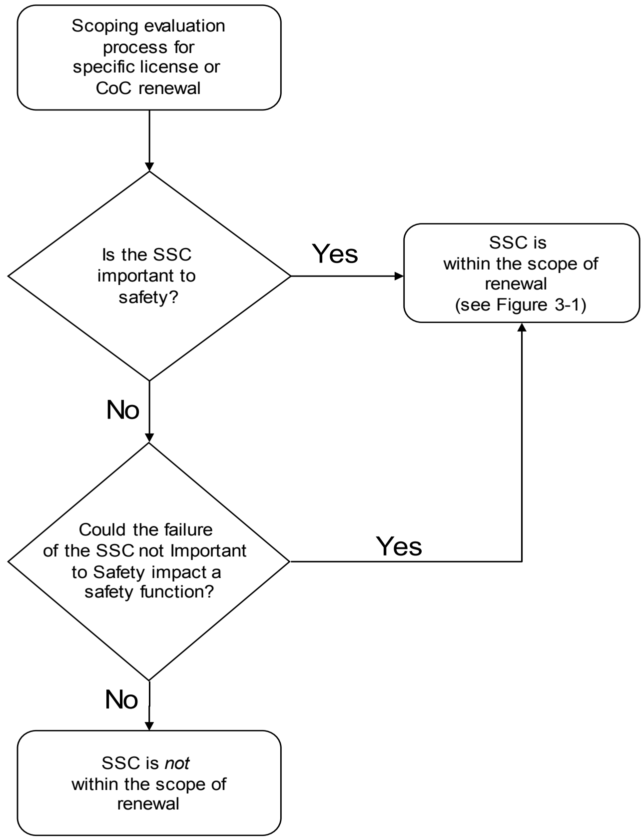

Figure 2-1. Flowchart of scoping evaluation process

The application should include a list and description of reference sources used to support the scoping evaluation. Sources may include the following:

- safety analysis reports (SARs), including final SARs (FSARs), updated FSARs, and topical SARs
- license or CoC
- technical specifications
- approved exemptions
- operating procedures
- design-bases documents (e.g., calculations, specifications, design change documents)
- drawings

- quality assurance plan or program
- docketed correspondence
- operating experience reports (site-specific or industrywide, as applicable)
- Title 10 of the *Code of Federal Regulations* (10 CFR) 72.48 (“Changes, Tests, and Experiments”) evaluations and screenings
- vendor information
- applicable NRC guidance

The reviewer can refer to NUREG/CR-6407, “Classification of Transportation Packaging and Dry Spent Fuel Storage System Components According to Importance to Safety,” as a reference for classification of components as important to safety to determine the accuracy and completeness of the scoping evaluation. The reviewer should ensure that the scoping evaluation has evaluated *all* SSCs identified in the design-bases documents and properly differentiated them as either within or *not* within the scope of renewal. In addition, the identification of SSCs and SSC subcomponents in the scoping evaluation should be consistent throughout the application.

2.4.2 Structures, Systems, and Components within the Scope of Renewal

The reviewer should verify that the SSCs (and associated subcomponents) within the scope of renewal fall into the following scoping categories:

(1) They are classified as important to safety, as they are relied on to do one of the following functions:

  i. maintain the conditions required by the regulations, specific license, or CoC to store spent fuel safely

  ii. prevent damage to the spent fuel during handling and storage

  iii. provide reasonable assurance that spent fuel can be received, handled, packaged, stored, and retrieved without undue risk to the health and safety of the public

  These SSCs ensure that important safety functions are met for (1) confinement, (2) radiation shielding, (3) sub-criticality control, (4) heat-removal capability, (5) structural integrity, and (6) retrievability.

(2) They are classified as <u>not</u> important to safety but, according to the design bases, their failure could prevent fulfillment of a function that is important to safety.

The reviewer should verify that SSCs within the scope of renewal are screened to identify and describe the subcomponents with intended functions. The reviewer should recognize that SSC subcomponents may degrade by different modes, or have different criteria for evaluation from the overall component (i.e., different materials or environments).

The scoping evaluation should clearly (1) define the intended function of each SSC subcomponent and (2) differentiate SSC subcomponents per scoping criteria 1 and 2, as defined above. The reviewer should ensure that this information is tabulated or adequately described in the application. The reviewer should confirm that this information is comprehensive and accurate (i.e., SSC subcomponents are not missing from the scoping evaluation; SSC subcomponent naming is consistent with the design bases; intended functions are properly described) by comparing the results of the scoping evaluation to appropriate final safety analysis report (FSAR) drawings or tables.

2.4.2.1 Scoping of Fuel Assemblies

Traditional light water reactor spent nuclear fuel consists of fuel rods and assembly hardware. In turn, the fuel rods consist of uranium oxide pellets inside a cladding tube. The spent fuel cladding and assembly hardware provide structural support to ensure that the spent fuel is maintained in a known geometric configuration. The safety analyses for an ISFSI or DSS (e.g., criticality and shielding analyses) may rely on the fuel assembly having a specific configuration (e.g., geometric form, a certain number of fuel rods or solid replacement filler rods in the assembly lattice). As the renewal of a specific license or CoC is based on continuation of the approved design bases throughout the period of extended operation (as discussed in Section 2.3), for these ISFSIs or DSSs, the renewal application should demonstrate that the analyzed fuel configuration is maintained during the period of extended operation. Therefore, the condition of the fuel assembly and cladding are within the scope of renewal and should be reviewed for any aging mechanisms and effects that may lead to a change in the analyzed fuel configuration. If a licensee or CoC holder wishes to revise the safety analyses for the approved design-bases fuel configuration (i.e., analyzed fuel configuration), it should pursue such a change through an amendment or revision request, and not as part of the renewal application.

2.4.2.2 Scoping of Structures, Systems, and Components, Depending on Individual Design Bases

In some cases, transfer casks, transporter devices, reinforced concrete pads, and other engineered features (e.g., earthen berms, shield walls, or engineered fill within an underground ISFSI) may be classified as important to safety or safety-related (under 10 CFR Part 50) in the design bases of various ISFSIs or DSSs. The reviewer should review the FSAR to determine how these SSCs are used in the FSAR evaluations and described in the license or CoC, to understand whether these SSCs are considered part of the design bases, and thus whether they are considered to be within the scope of renewal.

2.4.3 Structures, Systems, and Components Not within the Scope of Renewal

For those SSCs (and associated subcomponents) excluded from the scope of renewal, the reviewer should verify that they do not meet either of the criteria described in Section 2.4.2. The reviewer should ensure that the applicant has properly justified any exclusions by referencing the design bases (i.e., FSAR description, drawings, or tables).

The following SSCs may be excluded from the scope of renewal, provided that they do not meet either of the criteria in Section 2.4.2 above:

- equipment associated with cask loading and unloading, such as (1) welding and sealing equipment, (2) lifting rigs and slings, (3) vacuum-drying equipment, (4) portable radiation

survey equipment, and (5) other tools, fittings, hoses, and gauges associated with cask loading and unloading

- Instrumentation and other active components/systems (i.e., not passive or long-lived, but subject to a change in configuration or replacement based on a qualified life or service time period)

- miscellaneous hardware that does not support or perform any function that is important to safety

2.5 Evaluation Findings

The reviewer prepares the summary statement and evaluation findings based on compliance with the regulatory requirements described in Section 2.3. The summary statement and evaluation findings should be similar in wording to the following example (the finding number is for convenience in cross-referencing within the Standard Review Plan (SRP) and safety evaluation report (SER)):

The NRC staff reviewed the scoping evaluation provided in the renewal application and supplemental documentation. The NRC staff performed its review following the guidance provided in NUREG-1927, Revision 1, "Standard Review Plan for Specific Licenses and Certificates of Compliance for Dry Storage of Spent Nuclear Fuel," and relevant ISGs. The NRC staff used the information provided in NUREG/CR-6407 ("Classification of Transportation Packaging and Dry Spent Fuel Storage System Components According to Importance to Safety") in its review as a reference for classification of components as important to safety to determine the accuracy and completeness of the scoping evaluation. Based on its review, the NRC staff finds:

F2.1 The applicant has identified all SSCs important to safety and SSCs failure of which could prevent an SSC from fulfilling its safety function, per the requirements of 10 CFR 72.3, 10 CFR 72.24, 10 CFR 72.42, 10 CFR 72.120, 10 CFR 72.122, 10 CFR 72.124, 10 CFR 72.126, 10 CFR 72.128 and 10 CFR 72.236, as applicable.

F2.2 The justification for any SSC determined not to be within the scope of the renewal is adequate and acceptable.

3. AGING MANAGEMENT REVIEW

3.1 Review Objective

The purpose of the aging management review (AMR) is to assess the proposed aging management activities (AMAs) for structures, systems, and components (SSCs) determined to be within the scope of renewal. The AMR addresses aging mechanisms and effects [1] that could adversely affect the ability of the SSCs (and associated subcomponents) from performing their intended functions during the period of extended operation. The reviewer should verify that the renewal application includes specific information that clearly describes the AMR performed on SSCs within the scope of renewal.

3.2 Areas of Review

The reviewer should ensure that the AMR in the renewal application provides the following content with adequate technical bases:

- identification of materials and environments for those SSCs and associated subcomponents determined to be within the scope of renewal

- identification of aging mechanisms and effects requiring management

- identification of time-limited aging analyses (TLAAs), if applicable, and aging management programs (AMPs) for managing the effects of aging

Figure 3-1 contains a flowchart for the AMR process. The final safety analysis report (FSAR) and supporting documents related to the design are the primary documents that describe the safety classification, intended function, materials, and service environments for SSCs of independent spent fuel storage installations (ISFSIs), dry storage systems (DSSs), or both, identified to be within the scope of renewal (see Section 2.4.1).

The reviewer should consult applicable consensus codes and standards that provide additional guidance on the applicability of aging mechanisms and effects. The use of ambiguous terminology from any standard (e.g., “change in material properties”) should be properly defined or referenced in the application. Refer to Appendix A for assessing non-quantifiable terms.

1 To effectively manage an aging effect, it is necessary to determine the aging mechanisms that are potentially at work for a given material and environment application. Therefore, the aging management review process identifies both the aging effects and the associated aging mechanisms that cause them.

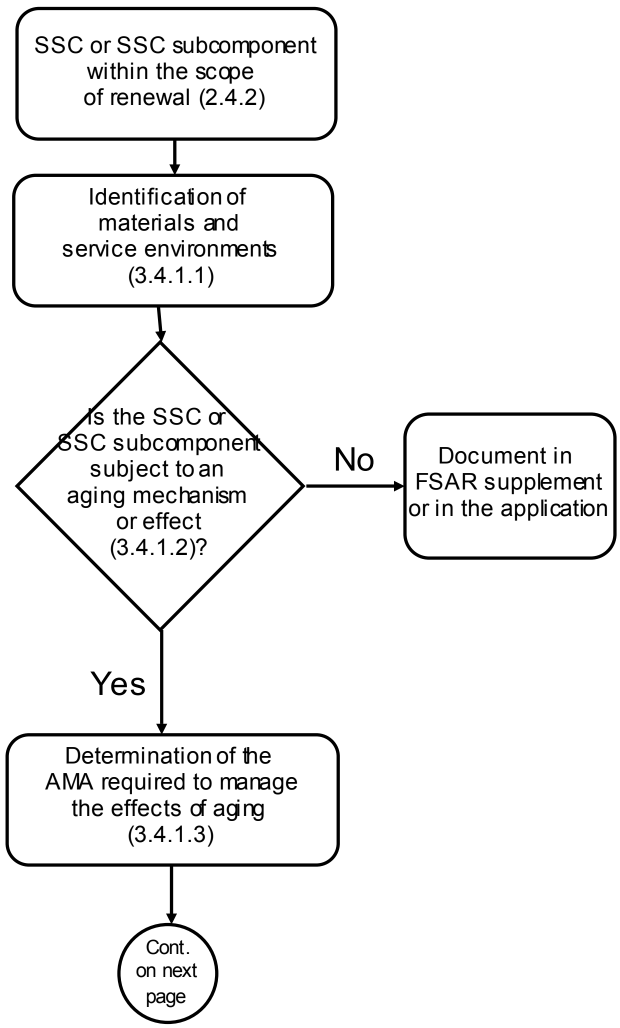

Figure 3-1. Flowchart of AMR process

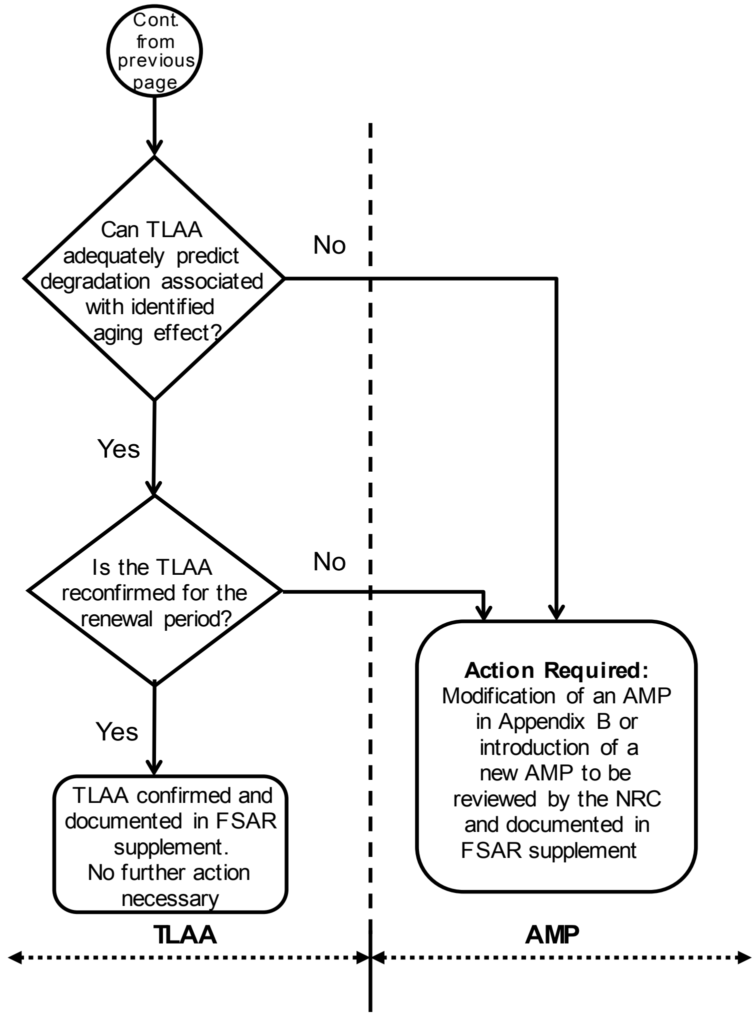

Figure 3-1. Flowchart of AMR process (continued)

3.3 Regulatory Requirements

Title 10 of the *Code of Federal Regulations* (10 CFR) 72.42 and 10 CFR 72.240 provide the overarching requirements for aging management activities for renewal of specific licenses and certificates of compliance (CoCs). Table 3-1 presents a matrix of regulatory requirements that must continue to be met to ensure that the intended functions of SSCs of ISFSIs and DSSs are maintained during the period of extended operation. Other parts of 10 CFR Part 72 may also apply.

Table 3-1. Relationship of Regulations and Aging Management Reviews

Areas of Review	**10 CFR Part 72 Regulations**				
	72.24 [a] (d)	72.82 [a] (d)	72.104 [a] (a)	72.106 [a] (b)	72.120 [a] (a),(d)
Aging Effects	●		●	●	●
Aging Management, Maintenance, or Surveillance Programs		●			
TLAAs	●		●	●	●

Areas of Review	**10 CFR Part 72 Regulations**				
	72.122 [a] (a),(b),(c),(h)(1), (h)(5),(l)	72.122 [a] (f),(h)(4),(i)	72.124	72.126[a]	72.128 [a] (a)
Aging Effects	●		●	●	
Aging Management, Maintenance, or Surveillance Programs		●		●	●
TLAAs	●	●	●	●	●

Areas of Review	**10 CFR Part 72 Regulations**				
	72.158	72.162	72.164	72.168 (a)	72.170
Aging Effects	●	●	●		
Aging Management, Maintenance, or Surveillance Programs	●	●	●	●	●
TLAAs					●

Table 3-1. Relationship of Regulations and Aging Management Reviews (continued)

Areas of Review	**10 CFR Part 72 Regulations**		
	72.172	72.236 [b] Applicable Sections	72.240(d) [b]
Aging Effects		●	●
Aging Management, Maintenance, or Surveillance Programs	●	●	●
TLAAs			

[a] These regulations apply only to specific-license renewals per 10 CFR 72.13.
[b] These regulations apply only to CoC renewals per 10 CFR 72.13.

3.4 Materials, Service Environments, Aging Mechanisms and Effects, and Aging Management Activities

3.4.1 Review Guidance

This section provides review guidance for the aging management review. Section 3.4.1.1 describes the identification of materials and their environments. Section 3.4.1.2 describes the identification of aging mechanisms and effects. Section 3.4.1.3 describes the review of aging management activities. Section 3.4.1.4 describes the aging management review for spent fuel assemblies.

3.4.1.1 Identification of Materials and Environments

The AMR process includes the identification of the materials of construction and the service environments for each SSC (or SSC subcomponent) within the scope of renewal. The identification of SSCs and SSC subcomponents in the AMR process should be consistent with the identification in the scoping evaluation. The reviewer should ensure that the renewal application has provided environmental data (and referenced the source of the data) such that the range of operating and service conditions of the SSCs can be determined. The pertinent environmental data is that which has a direct bearing on aging and the proposed aging management approach, and may include:

- temperature
- wind
- relative humidity
- relevant atmospheric pollutants and deposits
- exposure to precipitation
- marine fog, salt, or water exposure
- radiation field (gamma and neutron)
- the service environment (e.g., embedded, sheltered, or outdoor)
- gas compositions (e.g., external: air; internal: inert gas such as helium)

The reviewer should verify that the applicant considered the specific environment and material combinations in the DSS design for the evaluation of observed and potential aging mechanisms and effects that may lead to a loss of intended function. Particular attention should be paid to dissimilar metal combinations that may result in galvanic effects such as hydrogen uptake or galvanic corrosion. For sheltered environments that are in contact with the site atmosphere, the potential deposition and accumulation of atmospheric deposits should be evaluated. For example, chloride salt deposits that can promote accelerated corrosion of SSCs may be relevant in locations near salt water, adjacent to cooling towers, or in close proximity to roads that are treated with deicing salts.

3.4.1.2 Identification of Aging Mechanisms and Effects

The reviewer should ensure that the applicant has provided an analysis and documentation that identify aging mechanisms and effects pertinent to the SSCs determined to be within the scope of renewal. The AMR should include aging mechanisms and effects that could reasonably be expected to occur, as well as those that have actually occurred, based on industry and site-specific operating experience and component testing. The reviewer should ensure that the CoC renewal evaluates all potential environments where the DSS may be used and provides applicable operating experience to justify the AMR conclusions.

The reviewer should review the applicant's synopses of information used to identify applicable aging mechanisms and effects. Identification of applicable aging mechanisms and effects may be through review of:

- site maintenance, repair, and modification records
- corrective action reports, including root cause evaluations
- pre-application inspection results (see below)
- maintenance and inspection records from ISFSI sites with similar SSC materials and operating environments
- industry records
- applicable operating experience outside the nuclear industry
- applicable consensus codes and standards
- U.S. Nuclear Regulatory Commission (NRC) reports
- other applicable guidance for determining if an aging mechanism or effect should be managed for the period of extended operation

Examples of potential aging mechanisms and effects that may be identified by reviewing the sources of information cited above include: (1) cracking or loss of strength as a result of cement aggregate reactions in the concrete, (2) cracking or loss of material as a result of freeze-thaw degradation of the concrete (requires the presence of moisture combined with temperatures below freezing), (3) reinforcement corrosion and concrete cracking as a result of chloride ingress, (4) accelerated corrosion of steel structures and components and stress corrosion cracking of austenitic stainless steels as a result of atmospheric deposition of chloride salts.

The reviewer should ensure that, if the applicant relies on operating experience, it is specifically applicable to the SSC subcomponent/material/environment, and is not just a compendium of references on similar topics.

The applicant is not required to take further action if an SSC is determined to be within the scope of renewal but is found to have no potential aging effects for the period of extended operation. The reviewer should verify that the applicant's exclusion of an aging mechanism or effect is consistent with maintenance records, operating experience, and information obtained during pre-application inspection(s). The reviewer should also ensure that the FSAR supplement or other application materials document the applicant's determination of SSCs requiring no further review.

Pre-Application Inspections

Because inspections of DSSs are not typically conducted during the initial storage period, or are narrow in scope, the reviewer may have limited information regarding applicable aging mechanisms and effects for the specific design, environment, and operational parameters. Although there is no specific regulatory requirement for a pre-application inspection, it is one means by which an applicant can demonstrate that an aging effect does or does not require management. It can provide valuable operating experience and can be used to verify the condition of SSCs and SSC subcomponents is as-expected. The pre-application inspection is performed before submittal of the specific-license or CoC renewal application, and the inspection results become part of the technical bases for renewal.

Pre-application inspections should bound the site conditions, system designs, material combinations, and operating parameters that may contribute to the potential aging mechanisms and effects for SSCs and SSC subcomponents within the scope of renewal. For example, if chloride-induced stress corrosion cracking (CISCC) of a stainless steel canister is a potential aging mechanism, then the pre-application inspection should be conducted on the canister that has the greatest susceptibility to CISCC. The determination of susceptibility may involve the initial heat load of the canister, expected temperature variations on the canister surface with priority given to the coolest locations and welds, and the canister location if it is determined that some DSSs at a site may be located closest to or oriented toward a source of atmospheric chlorides. To address such variables, particularly for CoC renewal applications, pre-application inspections may involve SSCs in multiple DSSs at an ISFSI, and DSSs at multiple ISFSIs, as applicable.

Pre-application inspections may not include transfer casks or other similar SSCs that are leased or otherwise not actually on site. The latter SSCs are generally subject to maintenance requirements before use. Records from these maintenance activities may be included in the application in support of their respective aging management programs. Applicants are encouraged to discuss their considerations for selecting the system(s) to inspect with NRC staff in pre-application meetings before submitting the renewal application.

The reviewer should ensure that the scope, methods, and acceptance criteria for pre-application inspections align with the guidance for the AMP elements described in Section 3.6.1 of this document. The reviewer may use technical information in the example aging management programs in Appendix B as a reference when reviewing the adequacy of pre-application inspections. The reviewer should ensure that the application provides a description of any initiated corrective actions (including results from actions to verify extent of condition) due to conditions identified in the pre-application inspection(s).

The reviewer may accept the use of surrogate inspections (inspections conducted at other sites as a substitute for inspections conducted at the site(s) within the subject license or CoC) for identifying the relevant aging mechanisms and effects in the renewal application, but only when the technical basis is supported by substantial operating experience. Differences in materials, fabrication practices, design modifications, and environmental conditions at various sites could make comparisons between different ISFSI sites invalid.

The CoC holder is responsible for providing the technical basis for the proposed approach to aging management at multiple sites (where the CoC is and can be used). Therefore, pre-application inspections are likely only practical at a subset of sites that use the CoC. In this case, the reviewer should ensure that the chosen subset of sites is bounding with respect to the susceptibility of the various potential aging effects. Although CoC holders do not have the authority to conduct pre-application inspections at general licensee ISFSI sites, CoC holders could work within their user groups to identify bounding systems for the pre-application inspections. If pre-application inspections are not performed for each of the sites with the subject license or CoC, the reviewer should ensure that aging management programs include baseline inspections at each site upon entering the period of extended operation.

During baseline inspections, or the first inspections conducted per the AMPs, each licensee should assess the condition of SSCs: (1) to confirm the results of pre-application inspections that were conducted at other sites are bounding, or (2) to verify the adequacy of the AMPs and the conclusions of the TLAAs, when pre-application inspections were not performed. Also, the reviewer should consider whether uncertainties in SSC degradation due to the lack of a pre-application inspection warrant conditions in the license or CoC to require baseline inspections immediately upon entering the period of extended operation (see Section 1.4.7).

3.4.1.3 Aging Management Activities

The reviewer should ensure that the applicant has identified those aging mechanisms and effects requiring either an AMP or TLAA. Figure 3-1 illustrates the process for handling those SSCs that are determined to be within the scope of renewal and subject to a potential aging effect. The AMR defines two methods for addressing potential aging mechanisms and effects: TLAA (Section 3.5) and AMP (Section 3.6).

The NRC may condition the approval of a renewal on the requirements of a given AMP being met during the period of extended operation (see Section 1.4.7). The CoC user (general licensee) would ordinarily carry out the activities described in this AMP. Under 10 CFR 72.212(b)(11) and 10 CFR 72.240(e), the NRC may add the appropriate condition(s) or technical specification(s) to the renewed CoC for general licensee implementation of the AMP. Specific licenses may also be similarly conditioned (see Section 1.4.7).

3.4.1.4 Aging Management Review for Fuel Assemblies

Because the DSS interior and cladding cannot be reasonably inspected, the reviewer should rely on lessons learned from NUREG/CR-6745, “Dry Cask Storage Characterization Project—Phase 1; CASTOR V/21 Cask Opening and Examination,” (Bare et al., 2001), and NUREG/CR-6831, “Examination of Spent PWR Fuel Rods after 15 Years in Dry Storage,” (Einziger et al., 2003). This research demonstrated that low burnup fuel cladding and other cask internals had no deleterious effects after 15 years of storage. This research confirmed the basis for the guidance on creep deformation in Interim Staff Guidance (ISG) 11, “Cladding Considerations for the Transportation and Storage of Spent Fuel,” Revision 3, (NRC 2003). The

NRC staff has indicated in ISG 11, Revision 3 that the spent fuel configuration is expected to be maintained as analyzed in the safety analyses for the ISFSI or DSS, provided certain acceptance criteria (regarding maximum fuel clad temperature and thermal cycling) are met, and the fuel is stored in a dry inert atmosphere. These research results suggest that degradation of low burnup fuel cladding and assembly hardware should not occur during the first renewal period, provided that the cask or canister internal environment is maintained.

The reviewer should assess whether the applicant has considered the most recent revision of ISG-11 and research results in this area, especially with respect to high burnup (HBU) fuel. Research into fuel performance in storage is an ongoing effort.[2] The reviewer should ensure that the applicant has monitored new research developments to ensure it has identified any new potential aging mechanisms and effects and provided new supporting data demonstrating HBU fuel performance during the period of extended operation. Although NRC has confidence based on short-term testing (i.e., laboratory scale testing up to a few months) that there is no degradation of HBU fuel that would result in spent fuel being in an unanalyzed configuration in the period of extended operation, there is no operational data to demonstrate this as was done in the aforementioned demonstration on low burnup fuel.

Guidance for one acceptable approach to demonstrate the HBU fuel performance during the period of extended operation is provided in the “Example of a High Burnup Fuel Monitoring and Assessment Program” in Appendix B. This is a licensee program that monitors and assesses data and other information regarding HBU fuel performance, to confirm that the analyzed HBU fuel configuration is maintained during the period of extended operation. Guidance for determining if a surrogate demonstration program can provide the data to support a licensee’s HBU Fuel Monitoring and Assessment Program is given in Appendix D. Alternative approaches to that presented in Appendix B may be used, as appropriately justified by an applicant.

3.4.2 Evaluation Findings

The reviewer prepares the summary statement and evaluation findings based on compliance with the regulatory requirements in Section 3.3. The summary statement and evaluation findings should be similar in wording to the following example (the finding number is for convenience in cross-referencing within the Standard Review Plan (SRP) and safety evaluation report (SER)):

The NRC staff reviewed the aging management review provided in the renewal application and supplemental documentation. The NRC staff performed its review following the guidance provided in NUREG-1927, Revision 1, “Standard Review Plan for Renewal of Specific Licenses and Certificates of Compliance of Dry Storage of Spent Nuclear Fuel,” and relevant ISGs. Based on its review, the NRC staff finds:

F3.1 The applicant’s AMR process to be comprehensive in identifying the materials of construction and associated operating environmental conditions for those SSCs within the scope of renewal and has provided a summary of the information in the renewal application and FSAR supplement.

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[2] For example, research programs at Argonne National Laboratory (for NRC and the U.S. Department of Energy (DOE)), and other programs performed for the Nuclear Regulation Authority (formerly Japan Nuclear Energy Safety Organization) have studied hydride reorientation effects; Oak Ridge National Laboratory has also studied bending responses of the fuel.

F3.2 The applicant's AMR process to be comprehensive in identifying all pertinent aging mechanisms and effects applicable to the SSCs within the scope of renewal and has provided a summary of the information in the renewal application and FSAR supplement.

3.5 Time-Limited Aging Analyses

TLAAs are calculations or analyses used to demonstrate that in-scope SSCs will maintain their intended function throughout an explicitly stated period of extended operation (e.g., 40 years). These calculations or analyses may be used to assess fatigue life (number of cycles to predicted failure), or time-limited life (operating timeframe until expected loss of intended function). TLAAs should account for environmental effects. Under 10 CFR 72.3, "Definitions," TLAAs are those calculations and analyses meeting all six of the following criteria:

(1) Involve SSCs important to safety within the scope of the specific-license renewal, as delineated in Subpart F of 10 CFR Part 72, or within the scope of the spent fuel storage CoC renewal, as delineated in Subpart L of 10 CFR Part 72, respectively.

(2) Consider the effects of aging.

(3) Involve time-limited assumptions defined by the current operating term.

(4) Were determined to be relevant by the specific licensee or certificate holder in making a safety determination.

(5) Involve conclusions or provide the basis of conclusions related to the capability of SSCs to perform their intended safety functions.

(6) Are contained or incorporated by reference in the design bases.

Under 10 CFR 72.42(a)(1) or 72.240(c)(2), the reviewer should ensure that the application includes a list of TLAAs. The NRC staff uses the FSAR and other documents where the design bases are detailed to perform the review and confirm that the applicant did not omit any TLAAs submitted as part of the approved design bases. The number and type of TLAAs vary depending on the design bases of the ISFSI or DSS. The reviewer should ensure that all six criteria set forth in 10 CFR 72.3 (and repeated in this section) are satisfied to conclude that a calculation or analysis is a TLAA.

The following examples illustrate analyses that are not TLAAs and need not be addressed under 10 CFR 72.42(a)(1) or 72.240(c)(2):

- Analyses with time-limited assumptions defined short of the initial license or CoC term, for example, an analysis for a component based on a service life that would not reach the end of the initial license or CoC term.

- Analyses *not* contained or incorporated by reference in the design bases. Although not TLAAs by definition, the reviewer should note that these analyses may be included in the renewal application to justify the proposed attributes of an AMP (e.g., inspection frequency, sample size) for a particular SSC within the scope of renewal. These analyses may further be used in the AMR to justify the exclusion of an aging mechanism/effect or SSC subcomponent from the scope of an AMP (i.e., not requiring

any aging management activities), if the analysis is approved by the staff and included or summarized in the application.

3.5.1 Review Guidance

The reviewer should ensure that the applicant has appropriately identified TLAAs by applying the six criteria described below for SSCs within the scope of renewal:

(1) *Involve SSCs important to safety within the scope of the specific-license or CoC renewal*. Chapter 2 of this SRP provides the reviewer guidance on the scoping methodology.

(2) *Consider the effects of aging*. The effects of aging include but are not limited to loss of material, change in dimension, change in material properties, loss of strength, settlement, and cracking. The reviewer should ensure that any calculations or analyses relying on environmental susceptibility criteria are adequately supported by a valid technical basis, such as NRC endorsed criteria or operating experience. An AMP might be more applicable in these cases.

(3) *Involve time-limited assumptions defined by the current operating term*. The defined operating term should be explicit in the analysis. Simply asserting that the SSC is designed for a DSS or ISFSI service life is not sufficient. Calculations, analyses, or testing that explicitly include a time limit should support the assertions.

(4) *Were determined to be relevant by the licensee or certificate holder in making a safety determination*. Relevancy is a determination that the applicant makes based on a review of the information available. A calculation or analysis is relevant if it can be shown to have a direct bearing on the action taken as a result of the analysis performed. Analyses are also relevant if they provide the basis for a safety determination, and, in the absence of analyses, the applicant might have reached a different conclusion.

(5) *Involve conclusions or provide the basis of conclusions related to the capability of SSCs to perform their intended safety functions*. The TLAA should provide conclusions or a basis for conclusions regarding the capability of the SSC to perform its intended function through the end of the period of extended operation. If the TLAA does not provide a conclusion supporting the capability of the SSC to perform its intended function through the end of the period of extended operation, then the TLAA is not confirmed, and the applicant should propose an AMP to address/manage the aging mechanism or effect on the SSC. Analyses that do not affect the intended functions of SSCs are not TLAAs.

(6) *Are contained or incorporated by reference in the design bases*. TLAAs should already be contained or incorporated by reference in the design-bases documents. Such documentation includes the (1) FSAR, (2) technical specifications, (3) correspondence to and from the NRC, (4) quality assurance plan, and (5) topical reports included as references in the FSAR. The reviewer should ensure that the applicant has provided any references cited in design-bases documents that may be needed to clarify the assumptions, methods, or values used in TLAAs.

The reviewer should ensure that the applicant has appropriately dispositioned an identified TLAA by one of the following methods:

- Demonstrate the existing analysis remains valid for the period of extended operation, has already considered the requested period of extended operation, and concludes that the SSC will continue to perform its intended function through the end of the requested period of extended operation.

- Revise or update the existing analysis to demonstrate that it has been projected to the end of the requested period of extended operation and concludes that the SSC will continue to perform its intended function through the end of the requested period of extended operation.

- Manage the effects of aging on the SSC for the requested period of extended operation through an AMP.

3.5.2 Evaluation Findings

The reviewer prepares the summary statement and evaluation finding based on compliance with the regulatory requirements in Section 3.3. The summary statement and evaluation finding should be similar in wording to the following example (the finding number is for convenience in cross-referencing within the SRP and SER):

The NRC staff reviewed the TLAAs provided in the renewal application and supplemental documentation. The NRC staff performed its review following the guidance provided in NUREG-1927, Revision 1, “Standard Review Plan for Renewal of Specific Licenses and Certificates of Compliance for Dry Storage of Spent Nuclear Fuel,” and relevant ISGs. The NRC staff verified that the TLAA assumptions, calculations, and analyses are adequate and bound the environment, and aging mechanism or aging effect for the pertinent SSCs. Based on its review, the NRC staff finds:

F3.3 The applicant identified all aging mechanisms and effects pertinent to SSCs within the scope of renewal that involve TLAAs. The methods and values of the input parameters for the applicant’s TLAAs are adequate. Therefore, the applicant’s TLAAs provide reasonable assurance that the SSCs will maintain their intended function(s) for the period of extended operation, require no further aging management activities, and meet the requirements in 10 CFR 72.42(a)(1) or 10 CFR 72.240(c)(2), as applicable.

3.6 Aging Management Programs

AMPs monitor and control the degradation of SSCs within the scope of renewal so that aging effects will not result in a loss of intended functions during the period of extended operation. An AMP includes all activities that are credited for managing aging mechanisms or effects for specific SSCs, including activities conducted during the initial storage period. An effective AMP prevents, mitigates, or detects the aging effects and provides for the prediction of the extent of the effects of aging and timely implementation of corrective actions before there is a loss of intended function.

AMPs should be informed, and enhanced when necessary, based on the ongoing review of both site-specific and industrywide operating experience, including relevant international and non-nuclear operating experience. Operating experience provides direct confirmation of the effectiveness of an AMP and critical feedback for the need for improvement. As new knowledge and data become available from new analyses, experiments, and operating experience,

licensees and CoC holders should revise existing AMPs (or pertinent procedures for AMP implementation) to address program improvements or aging issues.

3.6.1 Review Guidance

An AMP should contain the following 10 elements:

(1) scope of program
(2) preventive actions
(3) parameters monitored or inspected
(4) detection of aging effects
(5) monitoring and trending
(6) acceptance criteria
(7) corrective actions
(8) confirmation process
(9) administrative controls
(10) operating experience

Review of the AMPs should include an assessment of the 10 program elements to verify their technical adequacy. In general, the reviewer should examine the details of these 10 elements for managing the aging mechanisms and effects identified by the aging management review (AMR) process. The reviewer should recognize that an applicant may develop AMPs following a different format or style. For such reviews, the NRC staff should ensure that sufficient detail (i.e., supporting technical bases) is provided in the alternate format in comparison with the 10 AMP elements of this guidance.

The reviewer should determine if the proposed AMP is adequate for managing the aging mechanisms and effects of the SSCs identified by the AMR. The following sections provide specific guidance for the review of each element of an AMP.

3.6.1.1 Scope of Program

The scope of the program should list the specific SSCs and subcomponents covered by the AMP and the intended functions to be maintained. In addition, the element should state the specific materials, environments, and aging mechanisms and effects to be managed. The reviewer should verify that the scope defined for the AMP is clear and specific.

3.6.1.2 Preventive Actions

Preventive actions are used to prevent aging or mitigate the rates of aging for SSCs through the activities in the AMP. The reviewer should verify that these activities, if applicable, are described. For example, an applicant may cite a ground dewatering system to ensure control of long-term settlement of structures or the continuance of inspections to ensure that air inlet/outlet vents are not blocked. Some condition or performance monitoring programs do not rely on preventive actions and thus this information need not be provided.

The reviewer should ensure that any proposed preventive action will not result in an unintended consequence to the ability of an SSC to fulfill its intended function(s). For example, if the applicant has proposed to change a coating system to prevent loss of material due to corrosion, the reviewer should ensure that the applicant has verified coating compatibility to confirm that the proposed action will not compromise the intended function(s) of the SSC.

3.6.1.3 Parameters Monitored or Inspected

This program element should identify the specific parameters that will be monitored or inspected and describe how those parameters will be capable of identifying degradation or potential degradation before a loss of intended function. The use of the parameters should be demonstrated to be capable of:

- monitoring the effectiveness of activities that prevent or mitigate aging (e.g., environmental controls)
- monitoring the performance of SSCs as an indirect indicator of degradation (e.g., radiation rate monitoring at the external surface of a cask)
- detecting, through direct inspection, the presence and severity of conditions or discontinuities that may have an effect on the function of SSCs (e.g., nondestructive examination of a component surface)

The reviewer should ensure that this program element provides a clear link between the aging effects identified in the scope of the program and the parameters monitored or inspected.

3.6.1.4 Detection of Aging Effects

Detection of aging effects should occur before there is a loss of intended function for any SSC identified within the scope of the program. This element should include inspection and monitoring details, including method or technique (i.e., visual, volumetric, surface inspection), frequency, sample size, data collection, and timing of inspections to ensure timely detection of aging effects. In general, the information in this element describes the "when," "where," and "how" of the AMP (i.e., the specific aspects of the activities to collect data as part of the inspection or monitoring activities).

The reviewer should ensure that the applicant has provided sufficient details on the following aspects of the inspection or monitoring activities:

- Method or technique: Consistent with quality assurance requirements, the method should be demonstrated to be capable of evaluating the condition of the SSC against the acceptance criteria for the specific aging mechanism or effect being monitored or inspected (as defined in AMP Element 6). For example, the applicant should provide a valid technical basis that a particular visual inspection method or instrument has sufficient resolution to identify a specific crack or defect dimension. Inspections should utilize consensus codes and standards, as applicable.
- Frequency: The reviewer should ensure that the proposed intervals for inspection or monitoring are consistent with applicable site-specific, design-specific, or industrywide operating experience. Inspections should have sufficient frequency to ensure that intended functions will be maintained until the next scheduled inspection.
- Sample size and selection of SSCs for inspection: For a limited sample size, the applicant should identify and justify the number of SSCs to be evaluated per inspection, including the extent of the inspection for each SSC (e.g., all accessible areas of five concrete overpacks in service), and criteria for selection of the specific SSCs for inspection based on parameters that may contribute to the operable aging mechanisms

and effects. Consideration should also be given to event-driven fabrication or operational issues that may contribute to degradation when selecting SSCs for inspection (e.g., welding repairs, occurrence of natural or man-induced events, exposure to potentially corrosive environments before the storage term, duration of time between fabrication of an SSC and the start of the storage term). The reviewer should ensure the applicant has justified using a limited sample size (e.g., one cask per pad) with a technical basis, which should include applicable site-specific, design-specific and industrywide operating experience. The application should also define the areas that have been determined to be inaccessible or below-grade and propose how the condition of the inaccessible SSCs will be assessed. The reviewer should ensure that the scope of each inspection is properly defined for both accessible and inaccessible (including below-grade) areas.

- Data collection: The application should reference any specific methods to be used for data acquisition and documentation, including any applicable consensus codes and standards. For example, the application may reference field evaluation guides for evaluating and documenting cracks in concrete (e.g., ACI 224.1R, ACI 201.1R).

- Timing of inspections: If pre-application inspections were not performed for each of the sites covered by or using the subject license or CoC, the reviewer should ensure that AMPs include baseline inspections at each site upon entering the period of extended operation. Baseline inspections, or the first inspections conducted within the AMPs, should assess the condition of SSCs to confirm the results of the pre-application inspections that were conducted at other sites or to verify the technical justification provided in the application when pre-application inspections were not performed. The reviewer should also consider any specific information on the proposed inspection schedule (e.g., time of the year) to support the effective detection of aging effects before a loss of intended function.

3.6.1.5 Monitoring and Trending

Monitoring and trending should provide for an evaluation of the extent of the effects of aging and the need for timely corrective or mitigative actions. This element describes how the data collected will be evaluated. This includes an evaluation of the results against the acceptance criteria and an evaluation regarding the rate of degradation to ensure that the timing of the next scheduled inspection will occur before a loss of intended function. For most cases, this element should have a baseline established before or at the beginning of the period of extended operation. The reviewer should determine if a baseline inspection is necessary to establish the parameters to be monitored and the trending analysis and, if so, whether the proposed baseline inspection is adequate.

Although the licensee may have the flexibility to inspect different SSCs or SSC subcomponents during subsequent inspections, the selection of different components for the inspections defined in the AMP relative to the components inspected in the baseline inspection may present issues for purposes of monitoring and trending. If subcomponents from different systems will be inspected over the period of extended operation, the monitoring and trending element should address how a given operable degradation mode will be adequately trended.

3.6.1.6 Acceptance Criteria

The acceptance criteria, against which the need for corrective action will be evaluated, should ensure that the SSC intended functions and the approved design bases are maintained during the period of extended operation. The proposed acceptance criteria should be appropriately justified.

The acceptance criteria could be specific numerical values or could consist of a discussion of the process for calculating specific numerical values of conditional acceptance criteria to ensure that the design bases are maintained. The reviewer should ensure that the acceptance criteria:

- Include a quantitative basis (justifiable by operating experience, engineering analysis, consensus codes/standards).
- Avoid use of non-quantifiable phrases (e.g., significant, moderate, minor, little, slight, few; see Appendix A).
- Are achievable and actionable.

The acceptance criteria may be taken directly from the design bases information included in either the final safety analysis report (FSAR) or technical specifications. The acceptance criteria also may be established by methods provided in NRC-approved topical reports or appropriate codes and standards.

3.6.1.7 Corrective Actions

Corrective actions are the measures to be taken when the acceptance criteria are not met. Timely corrective actions, including root cause determination and prevention of recurrence for significant conditions adverse to quality, are critical for maintaining the intended functions of the SSCs during the initial storage period as well as the period of extended operation.

Corrective actions should be described in adequate detail or referenced to source documents. An applicant may reference the use of a Corrective Action Program (CAP) approved under 10 CFR Part 50, Appendix B, “Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants,” or 10 CFR Part 72, Subpart G, “Quality Assurance,” to capture and address aging effects identified in the period of extended operation. At a minimum, all conditions that do not meet the AMP acceptance criteria should be entered into the CAP. The QA Program ensures that corrective actions are completed within the specific or general licensee’s CAP and includes provisions (as applicable) to:

- Perform functionality assessments.
- Perform apparent cause evaluations and root cause evaluations.
- Address the extent of condition.
- Determine actions to prevent recurrence for significant conditions adverse to quality.
- Justify non-repairs.

- Ensure corrective actions are adequately and effectively performed and do not have an adverse effect (aging-related or otherwise) on the subject component or other SSCs.

- Trend conditions.

The CAP should be able to respond to and adequately address any ISFSI or DSS aging issues. Also, the CAP's response to addressing any ISFSI or DSS aging issues should include any specific corrective actions specified in the license or CoC renewal application.

In some cases, the reviewer may determine the need for specific corrective actions, rather than referring only to the use of the CAP. For example, when very limited information exists on the condition of an SSC or the applicability of an aging effect at the time of the application, the application should include specific follow-up activities when AMP acceptance criteria are not met. Thus, the reviewer should review the corrective actions element with consideration of the safety-significance of the SSC and the ISFSI site-specific, DSS design-specific, and industrywide operating experience to ensure that the proposed corrective actions are adequate and effective.

3.6.1.8 Confirmation Process

This element of the program is intended to verify that preventive actions are adequate and that appropriate corrective actions have been completed and are effective. The confirmation process is commensurate with the specific or general licensee QA Program approved under 10 CFR Part 72, Subpart G, or 10 CFR Part 50, Appendix B. The QA Program ensures that the confirmation process includes provisions to preclude repetition of significant conditions adverse to quality. The reviewer should ensure the confirmation process describes or references procedures to:

- determine follow-up actions to verify effective implementation of corrective actions
- monitor for adverse trends due to recurring or repetitive findings or observations

The reviewer should be aware that the effectiveness of prevention and mitigation programs should be verified periodically. For example, in managing corrosion of carbon steel components, a mitigation program (coating) may be used to minimize susceptibility to corrosion. However, it also may be necessary to have a condition monitoring program (visual or other types of inspections) to verify that corrosion is being prevented or controlled to prevent a loss of intended function.

3.6.1.9 Administrative Controls

Administrative controls provide a formal review and approval process. Thus, any aging management programs must be administratively controlled and included in the FSAR supplement. The administrative controls are in accordance with the specific or general licensee QA Program approved under 10 CFR Part 72, Subpart G or 10 CFR Part 50, Appendix B, respectively. The QA Program ensures that the administrative controls include provisions, such as those that define:

- instrument calibration and maintenance
- inspector requirements
- record retention requirements
- document control

3.6.1.10 Operating Experience

The reviewer should verify that the operating experience element of the program supports a determination that the effects of aging will be adequately managed so that the SSC intended functions will be maintained during the period of extended operation. Operating experience is useful in providing justification for the effectiveness of each AMP program element and critical feedback for enhancement. The reviewer should verify that any degradation in the referenced operating experience has been clearly identified as either age-related or event-driven, with proper justification for that assessment.

The reviewer should verify that the AMP references and evaluates applicable operating experience, including, but not necessarily limited to:

- internal and industrywide condition reports
- relevant international and non-nuclear operating experience
- pre-application inspection results (see below)
- licensee event reports
- vendor-issued safety bulletins
- NRC Generic Communications
- updated consensus codes, standards, or guides
- applicable industry-initiatives (e.g., Department of Energy or Electric Power Research Institute sponsored inspections)

The reviewer should consider operating experience of existing programs, including past corrective actions resulting in program enhancements or additional programs. A past failure would provide feedback from operating experience and should have resulted in appropriate program enhancements or new programs. This information can demonstrate where an existing program has been implemented correctly and where it has shortcomings in detecting degradation in a timely manner. This information should provide objective evidence to support or refute the conclusion that the effects of aging will be managed adequately so that the SSC intended functions will be maintained during the period of extended operation.

Pre-Application Inspections

Because inspections of DSSs and ISFSI SSCs are not typically conducted during the initial storage period, or are narrow in scope, the reviewer may have limited information regarding the extent and rate of operable degradation mechanisms for the specific design, environment, and operational parameters. In this case, the reviewer should ensure that the scope, methods, frequencies and acceptance criteria for monitoring and inspection activities in the aging management programs are sufficiently conservative, such that a loss of SSC-intended functions does not occur during the period of extended operation. Pre-application inspections are one means by which an applicant can provide operating experience to justify the specific information in the 10 elements of the proposed aging management programs. In addition, pre-application inspections may provide data to support a TLAA or other analysis that justifies that an aging

effect will not challenge a SSC's ability to perform its function(s) (see Section 3.5). The staff considers the results of a pre-application inspection, in conjunction with other relevant operating experience in its review of the renewal application.

Section 3.4.1.2 discusses the acceptable characteristics of a pre-application inspection, including the selection of bounding systems, the use of surrogate inspections, and ensuring that the inspection scope, methods, and acceptance criteria align with the guidance for the AMP elements described in Section 3.6.1 of this document. Section 3.4.1.2 also discusses the use of baseline inspections in an AMP upon entering the period of extended operation to assess the condition of SSCs at sites that were not the subject of a pre-application inspection. Furthermore, when pre-application inspections are not performed, the reviewer should consider whether uncertainties in the degradation of SSCs warrant the addition of conditions to the license or CoC to require the baseline and other specific inspections to ensure the AMPs are adequate, including for all potential user sites in the case of a renewed CoC (see Section 1.4.7).

Learning AMPs

The reviewer should ensure that the application includes provisions to conduct future reviews of site-specific, design-specific, and industrywide operating experience, including relevant international and non-nuclear operating experience, to confirm the effectiveness of the AMPs or identify a need to enhance or modify an AMP. The reviewer should verify that the applicant: (1) references a specific system to be used to obtain, aggregate, and enter site-specific, design-specific, and industrywide operating experience (e.g., Institute of Nuclear Power Operations database), and (2) has discussed how it intends to provide timely reporting of operating experience to this system.

The commitment to future reviews should ensure that, as knowledge and data become available from new analyses, experiments, and operating experience, the licensees and CoC holders will revise the existing AMPs (or pertinent procedures for AMP implementation) as necessary to address any lessons learned identified during the review of operating experience.

If an applicant follows this approach, the reviewer should ensure that the description of the periodic assessments includes specific performance criteria (e.g., program-specific performance indicators for each of the 10 AMP elements) and proposed actions based on the assessment findings. The reviewer should also ensure that the timing of the assessments appropriately considers the rate of aging degradation and the anticipated availability of data from industry initiatives. The reviewer should consider the frequency, acceptance criteria, and proposed corrective actions for these assessments for the requisite finding of reasonable assurance.

Nuclear Energy Institute (NEI) 14-03, "Format, Content, and Implementation Guidance for Dry Cask Storage Operations-Based Aging Management," Revision 1, provides a proposed framework for learning AMPs through the use of "tollgates." NEI 14-03 defines "tollgates" as periodic points within the period of extended operation when licensees would be required to evaluate aggregate feedback and perform and document a safety assessment that confirms the safe storage of spent fuel. Tollgates are described as an additional set of in-service assessments beyond the normal continual assessment of operating experience, research, monitoring, and inspections on DSS component and ISFSI SSC performance that is part of normal ISFSI operations for licensees during the initial storage period as well as the period of extended operation. The reviewer should be aware that an applicant may reference the use of "tollgates" in the renewal application. The reviewer should (1) assess the applicant's proposed periodic assessments of operating experience and other relevant data, and (2) make a

determination regarding the ability of these assessments to ensure the continued effectiveness of AMPs. The reviewer should ensure that tollgates determined to be necessary to demonstrate the continued effectiveness of the AMPs are included in the application.

NEI 14-03, Revision 1, also describes a framework for the aggregation and dissemination of operating experience across the industry through the use of an aging-related operating experience "clearinghouse," titled the ISFSI Aging Management Institute of Nuclear Power Operations Database (ISFSI AMID). Whether the applicant references the ISFSI AMID described in NEI 14-03 or proposes an alternative means to seek out operating experience, the reviewer should ensure that the application describes how industrywide operating experience and results of industry initiatives will be accessed, utilized, and assessed to ensure that AMPs are modified as appropriate.

At the time of publication, the staff was continuing its review of NEI 14-03, Revision 1, for proposed NRC endorsement, as discussed in the Introduction.

The NRC will inspect licensees' implementation of AMPs in the period of extended operation, including any licensee actions taken as part of the "learning" aspect of AMPs. The NRC will inspect licensees' periodic assessments of AMP effectiveness and any adjustments licensees have made to AMPs to respond to operating experience and ensure AMPs are effective for addressing aging effects in the period of extended operation.

3.6.2 Commencement of AMP(s) for CoC Renewals

An AMP for a renewed CoC commences at the end of the initial storage period for each loaded DSS (see Appendix F for discussion of storage terms, including an explanation of when storage begins). Activities in an AMP may start at different timeframes (e.g., an AMP can be implemented before the period of extended operation to capture baseline data and AMPs may have different frequencies of implementation). Additional considerations for CoC renewals and general licensee implementation of AMPs are provided in Appendix E.

3.6.3 Implementation of AMP(s)

Generally, licensees should develop the infrastructure for AMP implementation (e.g., procure equipment or contracts, train personnel, or update, revise, or develop procedures for implementing AMP activities) before entering the period of extended operation. However, this may not be possible when the initial storage term ends shortly after the license or CoC is renewed or if a license or CoC is in the period of timely renewal (Section 1.4.5). In such cases, the development of the infrastructure for AMP implementation generally should be no later than one year from the date the NRC issues a renewed specific license or CoC. The reviewer should ensure that the timing of implementation for each AMP is addressed in the application in a clear manner and is appropriately justified if it exceeds the above guidance. Additional considerations for CoC renewals and general licensee implementation of AMPs are provided in Appendix E.

3.6.4 Evaluation Findings

The reviewer prepares the summary statement and evaluation finding based on compliance with the regulatory requirements in Section 3.3. The summary statement and evaluation finding should be similar in wording to the following example (the finding number is for convenience in cross-referencing within the SRP and SER):

The NRC staff reviewed the aging management programs provided in the renewal application and supplemental documentation. The NRC staff performed its review following the guidance provided in NUREG-1927, Revision 1, "Standard Review Plan for Renewal of Specific Licenses and Certificates of Compliance for Dry Storage of Spent Nuclear Fuel" and relevant ISGs. Based on its review, the NRC staff finds:

F3.4 The applicant has identified programs that provide reasonable assurance that aging effects will be managed effectively during the period of extended operation, in accordance with 10 CFR 72.42(a)(2) or 10 CFR 72.240(c)(3), as applicable.

4. CONSOLIDATED REFERENCES

Title 10 of the *Code of Federal Regulations* (10 CFR) Part 2, "Agency Rules of Practice and Procedure," Washington, DC.

10 CFR Part 20, "Standards for Protection against Radiation," Washington, DC.

10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," Washington, DC.

10 CFR Part 51, "Environmental Protection Regulations for Domestic Licensing and Related Regulatory Functions," Washington, DC.

10 CFR Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants," Washington, DC.

10 CFR Part 72, "Licensing Requirements for the Independent Storage of Spent Nuclear Fuel, High-Level Radioactive Waste, and Reactor-Related Greater than Class C Waste," Washington, DC.

American Concrete Institute (ACI) 349.3R-02, "Evaluation of Existing Nuclear Safety-Related Concrete Structures" (Reapp 2010).

ACI 318-05, "Building Code Requirements for Structural Concrete and Commentary," 2005.

ACI 349-06, "Code Requirements for Nuclear Safety-Related Concrete Structures and Commentary," 2007.

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Dominion Resources, Inc., "North Anna ISFSI TN-32 High Burnup Dry Storage Research Project," May 13, 2015, ADAMS Accession No. ML15133A082.

Einziger, R.E., et al., "Examination of Spent PWR Fuel Rods after 15 Years in Dry Storage," NUREG/CR-6831, Argonne National Laboratory, Argonne, IL., 2003, ADAMS Accession No. ML032731021.

Electric Power Research Institute (EPRI)/U.S. Department of Energy (DOE), HBU Dry Storage Cask Research and Development Project Final Test Plan, February 27, 2014, DOE Contract No. DE-NE-0000593. Accessible at http://www.energy.gov/sites/prod/files/2014/03/f8/HBUDry%20StrgeCaskRDfinalDemoTestPlanRev9.pdf.

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NRC,[3] "Standard Review Plan for Spent Fuel Dry Storage Facilities," NUREG-1567, Rev. 0, Washington, DC, 2000, ADAMS Accession No. ML003686776.

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NRC, "Standard Review Plan for Renewal of Spent Fuel Dry Cask Storage System Licenses and Certificates of Compliance," NUREG-1927, Rev. 0, Washington, DC, 2011, ADAMS Accession No. ML111020115.

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NRC, "Potential Chloride-Induced Stress Corrosion Cracking of Austenitic Stainless Steel and Maintenance of Dry Cask Storage System Canisters," Information Notice 2012-20, 2012b, ADAMS Accession No. ML12319A440.

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5. GLOSSARY

Accident condition: The extreme level of an event or condition, which has a specified resistance, limit of response, and requirement for a given level of continuing capability, which exceeds off-normal events or conditions. Accident conditions include both design-basis accidents and conditions caused by natural and manmade phenomena.

Aging effect: The manifestation of an aging mechanism (e.g., cracking, loss of fracture toughness, loss of material).

Aging management activity (AMA): An application of either the aging management program (AMP) or time-limited aging analyses (TLAAs) to provide reasonable assurance that the intended functions of structures, systems, and components (SSCs) of independent spent fuel storage installations (ISFSIs) and dry storage systems (DSSs) are maintained during the period of extended operation.

Aging management program (AMP): A program for addressing aging effects that may include prevention, mitigation, condition monitoring, and performance monitoring. See Title 10 of the *Code of Federal Regulations* (10 CFR) 72.3, "Definitions."

Aging management review (AMR): An assessment conducted by the licensee or certificate of compliance (CoC) holder that addresses aging mechanisms and effects that could adversely affect the ability of SSCs from performing their intended functions during the period of extended operation.

Aging mechanism: The degradation process for a given material and environment which results in an aging effect (e.g., freeze-thaw degradation, neutron irradiation, erosion).

Amendment of a license or CoC: An application for amendment of a license or a CoC must be submitted whenever a holder of a specific license or CoC desires to amend the license or CoC (including a change to the license or CoC conditions). The application must fully describe the changes desired and the reasons for such changes, and following as far as applicable the form prescribed for original applications. See 10 CFR 72.56, "Application for Amendment of License," and 10 CFR 72.244, "Application for Amendment of a Certificate of Compliance".

Baseline inspection: The first inspection of an AMP to assess the condition of SSCs to either: (1) confirm that the results of pre-application inspections conducted at other sites are bounding of the subject site, or (2) verify the adequacy of the AMPs and the conclusions of the TLAAs when pre-application inspections were not performed.

Burnup: The measure of thermal power produced in a specific amount of nuclear fuel through fission, usually expressed in GWd/MTU (gigawatt days per metric ton uranium).

Can for damaged fuel: A metal enclosure that is sized to confine one damaged spent fuel assembly. A fuel can for damaged spent fuel with damaged spent-fuel assembly contents must satisfy fuel-specific and system-related functions for undamaged spent nuclear fuel (SNF) required by the applicable regulations.

Canister (in a dry storage system for SNF): A metal cylinder that is sealed at both ends and may be used to perform the function of confinement. Typically, a separate overpack performs the radiological shielding and physical protection function. See NUREG-1536, "Standard Review Plan for Spent Fuel Dry Storage Systems at a General License Facility."

Certificate of compliance (CoC) (for a dry storage system for SNF): The certificate issued by the NRC that approves the design of a spent fuel storage cask in accordance with the provisions of 10 CFR Part 72, "Licensing Requirements for the Independent Storage of Spent Nuclear Fuel, High-Level Radioactive Waste, and Reactor-Related Greater than Class C Waste," Subpart L, "Approval of Spent Fuel Storage Casks." See 10 CFR 72.3.

Certificate of compliance holder (CoC holder): A person who has been issued a CoC by the U.S. Nuclear Regulatory Commission (NRC) for a spent fuel storage cask design under 10 CFR Part 72. See 10 CFR 72.3.

Certificate of compliance user (CoC user): The general licensee that has loaded, or plans to load, a dry storage system (DSS) in accordance with a CoC issued under 10 CFR Part 72.

Confinement (in a dry storage system for spent nuclear fuel): The ability to limit or prevent the release of radioactive substances into the environment.

Confinement systems: Those systems, including ventilation, that act as barriers between areas containing radioactive substances and the environment. See 10 CFR 72.3.

Controlled area: The area immediately surrounding an ISFSI for which the licensee exercises authority over its use and within which it performs ISFSI operations. See 10 CFR 72.3.

Criticality: The condition wherein a system or medium is capable of sustaining a nuclear chain reaction.

Degradation: Any change in the properties of a material that adversely affects the performance of that material; adverse alteration. See NUREG-1536.

Design bases:[4] Information that identifies the specific function(s) to be performed by SSCs (both important-to-safety and not important-to-safety) of a facility or of a spent fuel storage cask and the specific values or ranges of values chosen for controlling parameters as reference bounds for design. These values may be (1) restraints, derived from generally accepted "state-of-the-art" practices for achieving functional goals, or (2) requirements, derived from analysis (based on calculation, experiments, or both) of the effects of a postulated event under which SSCs must meet their functional goals. See 10 CFR 72.3.

Dry storage: The storage of spent nuclear fuel in a DSS, which typically involves drying the DSS cavity and backfilling with an inert gas.

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[4] The NRC has removed references to either "current licensing basis" (CLB) or "licensing basis", and replaced them with "design bases" in this guidance revision. Neither "current licensing basis" nor "licensing basis" is defined in 10 CFR Part 72. In addition, in the Statement of Considerations for the 2011 Part 72 rulemaking change (NRC, 2011), the NRC stated "The NRC does not believe that it is appropriate for the CLB to be applied to cask CoC renewals, which are generic."

Dry storage system (DSS): A system that typically uses a cask or canister in an overpack as a component in which to store spent nuclear fuel in a dry environment. A DSS provides confinement, radiological shielding, sub-criticality control, structural support, and passive cooling of its spent nuclear fuel during normal, off-normal, and accident conditions.

General license: Authorizes the storage of spent fuel in an ISFSI at power reactor sites to persons (i.e., general licensee) authorized to possess or operate nuclear power reactors under 10 CFR Part 50 ("Domestic Licensing of Production and Utilization Facilities") or Part 52 ("Licenses, Certifications, and Approvals for Nuclear Power Plants"). The general license is limited to (1) that spent fuel which the general licensee is authorized to possess at the site under the specific Part 50 or Part 52 license for the site, and (2) storage of spent fuel in casks approved under the provisions of 10 CFR Part 72, Subpart L. See 10 CFR 72.210 ("General License Issued") and 72.212(a)(1)–(2).

High burnup (HBU) fuel: Spent nuclear fuel with burnups generally exceeding 45 GWd/MTU.

Horizontal storage module: A reinforced, heavy-walled concrete structure designed to store dry spent fuel canisters in a horizontal position at an independent spent fuel storage installation. The horizontal storage module provides physical protection of canisters and radiological shielding, while allowing passive cooling.

Important to safety (ITS): See structures, systems, and components (SSCs) important to safety.

Independent spent fuel storage installation (ISFSI): A complex designed and constructed for the interim storage of spent nuclear fuel, solid reactor-related greater-than-Class-C (GTCC) waste, and other radioactive materials associated with spent fuel and reactor-related GTCC waste storage. See 10 CFR 72.3.

Inspection: The examination of an SSC, using a nondestructive testing technique, to determine its current condition and if there is any damage, defect, or degradation that could have an adverse effect on the function of that SSC.

Intended function: A design-bases function defined as either (1) important to safety or (2) failure of which could impact a safety function.

Interim staff guidance (ISG): Supplemental information that clarifies important aspects of regulatory requirements. An ISG provides review guidance to NRC staff in a timely manner until standard review plans are revised accordingly. See NUREG-1536.

Monitored retrievable storage installation (MRS): A complex designed, constructed, and operated by the U.S. Department of Energy for the receipt, transfer, handling, packaging, possession, safeguarding, and storage of spent nuclear fuel aged for at least 1 year, solidified high-level radioactive waste resulting from civilian nuclear activities, and solid reactor-related GTCC waste, pending shipment to a HLW repository or other disposal. See 10 CFR 72.3.

Monitoring: Data collection (from activities performed in either the initial storage period or the period of extended operation) to determine the status of a DSS, ISFSI, or both, and to verify the continued efficacy of the system, on the basis of measurements of specified parameters, including temperature, direct radiation, radioactive effluents, functionality, and characteristics of components of the system. Monitoring could thus be described as those activities that periodically or continuously monitor performance as an indirect indicator of degradation (e.g., monitoring groundwater chemistry) or monitor the effectiveness of preventive measures. With respect to direct radiation and radioactive effluents, according to 10 CFR 20.1003, "Definitions," monitoring means the measurement of radiation levels, concentrations, surface area concentrations or quantities of radioactive material, and the use of the results of these measurements to evaluate potential exposures and doses. See NUREG-1536.

Normal events or conditions: The maximum level of an event or condition expected to routinely occur. Events and conditions that exceed the levels associated with "normal" are considered to be, and to have the response allowed for, "off-normal" or "accident-level" events and conditions. See NUREG-1567, "Standard Review Plan for Spent Fuel Dry Storage Facilities."

Off-normal events or conditions: The maximum level of an event or condition that, although not occurring regularly, can be expected to occur with moderate frequency (once per calendar year) and for which there is a corresponding maximum specified resistance, specified limit of response, or requirement for a specified level of continuing capability. Off-normal is considered to include "anticipated occurrences" as used in 10 CFR Part 72. See NUREG-1536.

Overpack: A heavy-walled concrete, metal, or combined concrete and metal structure designed to store spent fuel canisters at an ISFSI. The overpack provides physical protection of canisters and radiological shielding, while allowing passive cooling.

Pre-application inspection: An inspection performed at the discretion of the licensee or CoC holder before submittal of the renewal application to provide operating experience to support the aging management review, proposed AMP activities, or evaluation of TLAAs.

Radiation shielding: ISFSI and DSS SSCs that are designed so that dry storage operations at an ISFSI meet the requirements of 10 CFR 72.126(a)(6) and 10 CFR 72.128(a)(2) and the requirements of 10 CFR 72.104(a) and 10 CFR 72.106(b), when both direct radiation and radioactive effluents are considered.

Renewal of a license or CoC: A certificate holder may apply for renewal of the design of a spent fuel storage cask for a term not to exceed 40 years. In the event that the certificate holder does not apply for a cask design renewal, any licensee using a spent fuel storage cask, a representative of the licensee, or another certificate holder may apply for a renewal of that cask design for a term not to exceed 40 years. See 10 CFR 72.240, "Conditions for Spent Fuel Storage Cask Renewal." Specific licenses may be renewed by the Commission at the expiration of the license term upon application by the licensee for a period not to exceed 40 years. See 10 CFR 72.42, "Duration of License; Renewal."

Retrievability: Storage systems must be designed to allow ready retrieval of spent fuel, high-level radioactive waste, and reactor-related GTCC waste for further processing or disposal. See 10 CFR 72.122(l). ISG-2 provides guidance on the fuel retrievability, including ready retrieval.

Safety analysis report (SAR): The document that a CoC holder, specific licensee, an applicant for a CoC, or an applicant for a specific license supplies to the NRC for evaluation. For specific-license renewals, the SAR must contain information required in 10 CFR 72.24, "Contents of Application; Technical Information." For CoC renewals, the SAR must meet the requirements of 10 CFR 72.240(b). The SAR provides references and drawings of the DSS, ISFSI, or both; details of construction; materials; and standards to which the SSC has been designed or fabricated. For clarification, SAR is a general term; while FSAR indicates the document that is submitted within 90 days after the issuance of the license or CoC that is based on the SAR in the license or CoC application and reflects any changes or applicant commitments developed during the license or CoC approval and/or hearing process. Both FSAR and updated final safety analysis report (UFSAR) are terms that are used to indicate the FSAR update that is required every 2 years. A specific licensee or CoC holder shall update the FSAR in accordance with 10 CFR 72.70 ("Safety Analysis Report Updating") or 10 CFR 72.248, ("Safety Analysis Report Updating") respectively.

Safety evaluation report (SER): The document that the NRC publishes at the completion of a licensing or certification review. The SER contains all of the NRC staff findings and conclusions from the licensing or certification review.

Safety function: A function defined as ITS. The ITS functions that structures, systems, and components are designed to maintain include:

- structural integrity
- content temperature control (i.e., heat-removal capability)
- radiation shielding
- confinement
- sub-criticality control
- retrievability

See NUREG-1536.

Service conditions: Conditions (e.g., time of service, temperatures, environmental conditions, radiation, and loading) that a component experiences during storage.

Specific license: A license for the receipt, handling, storage, and transfer of spent fuel, high-level radioactive waste, or reactor-related GTCC waste that is issued to a named person (i.e., specific licensee) on an application filed under regulations in 10 CFR Part 72, Subpart B, "License Application, Form, and Contents."

Spent fuel storage cask or cask: All the components and systems associated with the container in which spent fuel, or other radioactive materials associated with spent fuel, is stored at an ISFSI. See 10 CFR 72.3.

Spent nuclear fuel or spent fuel: Nuclear fuel that has been withdrawn from a nuclear reactor after irradiation, has undergone at least a 1-year decay process since being used as a source of energy in a power reactor, and has not been chemically separated into its constituent elements by reprocessing. Spent fuel includes the special nuclear material, byproduct material, source material, and other radioactive materials associated with fuel assemblies. See 10 CFR 72.3.

Structures, systems, and components (SSCs) important to safety: See 10 CFR 72.3. Those features of the ISFSI and spent fuel storage cask whose functions are at least one of the following:

- to maintain the conditions required to safely store spent fuel, high-level radioactive waste, or reactor-related GTCC waste
- to prevent damage to the spent fuel, the high-level radioactive waste, or reactor-related GTCC waste container during handling and storage
- to provide reasonable assurance that spent fuel, high-level radioactive waste, or reactor-related GTCC waste can be received, handled, packaged, stored, and retrieved without undue risk to the health and safety of the public

Time-limited aging analysis (TLAA): See 10 CFR 72.3. A licensee or CoC holder calculation or analysis that has all of the following attributes:

- involves SSCs important to safety within the scope of license or CoC renewal
- considers the effects of aging
- involves time-limited assumptions defined by the current operating term, for example, 40 years
- was determined to be relevant by the licensee or CoC holder in making a safety determination
- involves conclusions or provides the basis for conclusions related to the capability of the SSCs to perform their intended safety functions
- is contained or incorporated by reference in the design bases

Transfer cask: A shielded SSC used to transfer the fuel canister between the spent fuel handling area and the overpack or storage module.

APPENDIX A

NON-QUANTIFIABLE TERMS

Appendix A

Non-Quantifiable Terms

It is preferred that renewal applications use quantifiable terms and quantitative information, where it exists. However, the following non-quantifiable terms, as well as others, may appear in the renewal application, safety analysis report (SAR), and updates to final SARs:

- large
- small
- slight
- slightly
- significant
- significance
- moderate
- moderately
- low
- minor
- many
- few
- little
- routine
- some
- major
- undetectable
- visible
- measurable
- unchanged
- changed
- no loss of

Table A-1 may be used as guidance for the terms listed above, for additional consideration, or to provide quantitative measures or information.

Table A-1. Screening Criteria for Non-Quantifiable Terms

	Terms	Actions
Screened In	**The term requires additional consideration if it is used for one of the following:** • characterizing an aging effect (e.g., degradation, cracking, fatigue, corrosion, loss of material, change in material properties) • providing important information about the operations, functions, or other characteristics of an in-scope SSC • describing dose, environmental impact, or other hazard, such as combustible material or dust	**If the term screens in, one of the following must be provided:** • quantitative information, if it is available • additional descriptions • definition of the meaning of the term (e.g., “insignificant” means the function of the SSC is not impaired)
Screened Out	**The term is considered immaterial to the SAR and ISFSI/DSS UFSAR for one of the following reasons:** • The term is included in the title of reference document. • The term is included in a quote. • The term is explained by adjacent quantitative information (e.g., small: less than 20 percent). • Use of the term is NOT related to any of the following: – in-scope SSCs per AMR results – aging effect – dose, environment impact, or other hazard (e.g., combustible material) • Use of the term does not provide important information. It is merely descriptive and the meaning of the statement is not changed if the term is deleted (e.g., the word “small” could be deleted from the following statement without altering the meaning: “Water in the grapple ring is drained through a small hole”).	No action

APPENDIX B

EXAMPLES OF AGING MANAGEMENT PROGRAMS

Appendix B

Examples of Aging Management Programs

Appendix B contains example aging management programs (AMPs) for:

- localized corrosion and stress corrosion cracking of welded stainless steel dry storage canisters (Table B-1)
- reinforced concrete structures (Table B-2)
- high burnup (HBU) fuel monitoring and assessment program (Table B-3)

This appendix provides examples of acceptable AMPs for staff reference during review of renewal applications.

Example AMP for Localized Corrosion and Stress Corrosion Cracking of Welded Stainless Steel Dry Storage Canisters

Welded stainless steel canisters are used in the majority of the dry storage systems in the United States for spent nuclear fuel from commercial power reactors at both specific-licensed and general-licensed independent spent fuel storage installations (ISFSIs). The welded stainless steel canisters are the primary confinement boundary during storage. Although there are no known operational occurrences of aging or localized corrosion of welded stainless steel canisters, operating experience with nuclear reactors that were located close to an open ocean or bay has shown that pitting corrosion, crevice corrosion, and chloride-induced stress corrosion cracking (CISCC) can occur in welded stainless steel components as a result of atmospheric deposition and deliquescence of chloride-containing salts. Laboratory and natural exposure tests suggest that CISCC can occur with sufficient surface chloride concentrations and that, with those concentrations of chloride, crack propagation rates can be of engineering significance for welded stainless steel canisters during the period of extended operation.

Based on reactor operating experience as well as laboratory and field testing, localized corrosion and CISCC are potential aging mechanisms for welded stainless steel canisters. Environments where chloride-containing salts may be deposited on welded stainless steel canisters include coastal locations near salt water and locations that are close to cooling towers or roads that are salted. ASME Section XI has formed a Task Group to develop a code case to establish the requirements for inservice inspection and acceptance criteria for dry storage system canisters. However, the development of a consensus based code case for inservice inspection of dry storage system canisters may take several years to complete. To address potential aging effects as a result of localized corrosion cracking and stress corrosion cracking in the absence of an acceptable code case, the U.S. Nuclear Regulatory Commission (NRC) has provided an example AMP for welded stainless steel canisters used in dry storage systems that relies on guidance from consensus codes and standards for inservice inspection of nuclear power plant components. Elements of an NRC staff developed example AMP are provided in Table B-1.

Table B-1. Example AMP for Localized Corrosion and Stress Corrosion Cracking of Welded Stainless Steel Dry Storage Canisters

Element	Description
1. Scope of Program	Inspection of welded stainless steel dry storage canister confinement boundary external surfaces for atmospheric deposits, localized corrosion, and stress corrosion cracking. Examinations should be focused on areas with the following attributes: • canister fabrication welds and weld heat affected zones • closure welds and weld heat affected zones • areas of the canister to which temporary supports or attachments were attached by welding and subsequently removed • locations where a crevice is formed on the canister surface • horizontal (±30°) surfaces where deposit accumulation may accumulate at a faster rate compared to vertical surfaces • canister surfaces that are cold relative to the average surface temperature • canister surfaces with higher amounts of atmospheric deposits Effort should be made to identify and prioritize examinations of areas on canisters that have two or more of the above attributes (e.g., canister surface that is cold relative to average surface temperature and also has a weld/weld heat affected zone).
2. Preventive Actions	None, AMP is for condition monitoring. However, dry storage system canister designs may include preventative actions, such as fabrication procedures and surface modification methods to impart compressive residual stresses on the canister welds and weld heat affected zones to reduce the potential for stress corrosion cracking. Preventative actions may also include the use of dry storage system canister confinement boundary materials that are resistant to localized corrosion and stress corrosion cracking. For such cases the preventative actions described should be supported with an analysis and data demonstrating the preventative actions are effective.
3. Parameters Monitored or Inspected	Parameters monitored/inspected should include: • visual evidence of discontinuities and imperfections, such as localized corrosion, including pitting corrosion, crevice corrosion and stress corrosion cracking of the canister welds and weld heat affected zones • size and location of localized corrosion and stress corrosion cracks • appearance and location of deposits on the canister surfaces

Element	Description
4. Detection of Aging Effects	Visual examination of deposits on the canister surfaces and to identify corrosion products that may be indicators of localized corrosion and stress corrosion cracking in the welds and weld heat affected zones. Visual examination instrumentation with demonstrated sizing and depth measurement capability may be useful in the determination of the size and depth of pits open to the surface. Visual examination may also detect the presence of cracks originating from pits. However, the ability to detect cracks on clean metal surfaces using visual examination methods is dependent on several factors and can be difficult for tight crack opening displacements (Cumblidge et al., 2004; 2007). The presence of significant corrosion product accumulation may also interfere with the identification of stress corrosion cracks using visual examination methods. Volumetric examination is necessary to characterize stress corrosion cracking. Volumetric examination of pits and areas immediately adjacent to pits is necessary when pits are within 25 mm (1 inch) of a through thickness weld or within 25 mm (1 inch) of an area where an temporary attachment was known to be located. <u>Visual Examination</u> Pitting and crevice corrosion that is open to the surface can potentially be detected by visual testing (ASME Section V, Table A-110). Because of the high neutron and gamma radiation fields near the surface of the stainless steel dry storage canisters, direct visual examination is not possible. Procedures for remote visual examination should be performance demonstrated; procedure attributes including equipment resolution, lighting requirements, etc., should reference applicable standards, such as ASME Section XI, Article IWA-2200 for VT-1 and VT-3 examinations (ASME, 2007) and BWRVIP-03 (Selby 2005) for EVT-1 examinations. <u>Volumetric Examination</u> Additional assessment is necessary for suspected areas of localized corrosion or stress corrosion cracking. In these cases, the severity of degradation must be assessed including the dimensions of the affected area and the depth of penetration with respect to the thickness of the canister. For accessible areas where adequate cleaning can be performed, remote visual examination meeting the requirements for VT-1 Examination (ASME Section XI, IWA-2211) may be used to determine the type of degradation present (e.g., pitting corrosion or stress corrosion cracking) and the location of degradation. Examinations to characterize the extent and severity of localized corrosion or stress corrosion cracking should be conducted using surface or volumetric examination methods consistent with the requirements of ASME Section XI, IWB-2500 for category B-J components (ASME, 2007).

Element	Description
4. Detection of Aging Effects (cont'd)	Sample Size For sites where inspections are necessary, a minimum of one canister at each site. Preference should be given to the canisters with the greatest susceptibility for localized corrosion or SCC. Factors to be considered include older and colder canisters with the greatest potential for the accumulation and deliquescence of deposited salts that may promote localized corrosion and stress corrosion cracking, types of systems used at site, canister location with respect to potential sources of atmospheric deposits, system design, and operating experience. Industry guidance on evaluating susceptibility has been published by EPRI (Fuhr et al., 2015). Justification for not conducting inspections for localized corrosion or stress corrosion cracking should be provided on a case-by-case basis for each ISFSI site where welded stainless steel canisters are in use. Acceptable justification may be based on a comparison of susceptibility for the ISFSI location versus at least two other ISFSI sites determined to have greater susceptibility but showed no evidence of localized corrosion or stress corrosion cracking in inspections completed within 5 years of the time of the assessment. The justification must consider the full range of available ISFSI susceptibility assessments and welded stainless steel canister examination results. Data Collection Canister Examination: Documentation of the examination of the canister, location, and appearance of deposits. Assessment of the suspect areas where corrosion products were observed as described in corrective actions. Bounding Analysis: A complete listing of other sites considered, susceptibility assessments for those sites and results of examinations conducted at those sites. Justification for not including other sites where examinations showed evidence of localized corrosion and/or stress corrosion cracking. Frequency Once every 5 years

Element	Description
4. Detection of Aging Effects (cont'd)	<u>Timing of Inspections</u> The timing of the inspections includes the pre-application inspection or general-licensee baseline inspection, performed per Sections 3.4.1.2 and 3.6.1.10 of this NUREG, and at the frequency specified by the AMP. Alternative detection methods or techniques may be provided. For these cases: • The method or technique should be adequate and proven to be capable of evaluating the condition of the external surface of the canister against the acceptance criteria for the detection of localized corrosion and stress corrosion cracking. • The proposed intervals for inspection or monitoring are consistent with applicable site-specific, design-specific, or industrywide operating experience and should have sufficient frequency to ensure that the confinement function will be maintained until the next scheduled inspection. • The data collection methods should be sufficient for evaluating localized corrosion and stress corrosion cracking and should reference specific methods to be used for data acquisition including any applicable consensus codes and standards.

5. Monitoring and Trending	Monitoring and trending methods are in accordance with ASME Section XI evaluation criteria.
	
	Monitoring and trending methods reference plans/procedures used to:
	• Establish a baseline before or at the beginning of the period of extended operation.
	• Track trending of parameters or effects not corrected following a previous inspection, including:
	  – the locations and size of any areas of localized corrosion or the stress corrosion cracking
	  – the disposition of canisters with identified aging effects and the results of supplemental canister inspections
	
	Monitoring and trending should also include:
	• documentation of the appearance of the canister, particularly at welds and in crevice locations, with images and video that will allow comparison in subsequent examinations
	• changes to the size and number of any rust colored stains as a result of iron contamination of the surface in subsequent inspections

<table>
<tr>
<td>6. Acceptance Criteria</td>
<td>
No indications of localized corrosion pits, etching, crevice corrosion, stress corrosion cracking, red-orange colored corrosion products emanating from crevice locations, or red-orange colored corrosion products in the vicinity of canister fabrication welds, closure welds, and welds associated with temporary attachments during canister fabrication.
Confirmed or suspected areas of crevice corrosion, pitting corrosion and stress corrosion cracking must be assessed in accordance with acceptance standards identified in ASME Section XI, IWB-3514. Flaws exceeding the acceptance standards in IWB-3514.1 must be evaluated using the acceptance criteria identified in IWB-3640.
<u>Indications Requiring Additional Evaluation</u>
Although shop and handling procedures include controls to prevent iron contamination of the stainless steel surfaces, contamination does occur and is usually identified by rust-colored surface deposits. Iron contamination can exacerbate CISCC in stainless steels. In accessible locations, removal of the deposits and rust stains that reveal undamaged welds (i.e., absence of pits, crack, localized attack, or etching) and the original machining/grinding marks on the stainless steel base metal, including weld heat affected zones, may be used to confirm that localized corrosion or stress corrosion cracking have not been initiated.
Indications of interest that are subject to additional examination and disposition include:

localized corrosion pits, crevice corrosion, stress corrosion cracking, and etching (note that these indications may be covered by obstructions (i.e., crevices)); deposits; or corrosion products
discrete red-orange colored corrosion products that are 1 mm in diameter or larger especially those adjacent to fabrication welds, closure welds, locations where temporary attachments may have been welded to and subsequently removed from the stainless steel dry storage canister, and the weld heat affected zones of these areas
linear appearance of any color of corrosion products of any size parallel to or traversing fabrication welds, closure welds, locations where temporary attachments may have been welded to and subsequently removed from the stainless steel dry storage canister, and the weld heat affected zones of these areas
red-orange colored corrosion products greater than 1 mm in diameter combined with deposit accumulations in any location of the stainless steel canister
red-orange colored corrosion tubercles of any size
red-orange corrosion products present at the mouth of a crevice that includes a portion of the canister surface

Alternative acceptance criteria may be provided. For such cases, the acceptance criteria should:

Include a quantitative basis (justifiable by operating experience, engineering analysis, consensus codes/standards).
Avoid use of non-quantifiable phrases (e.g., significant, moderate, minor, little, slight, few).
Be achievable and clearly actionable.

</td>
</tr>
</table>

Element	Description
7. Corrective Actions	The corrective actions are in accordance with the specific or general licensee Quality Assurance (QA) Program approved under 10 CFR Part 72, Subpart G, or 10 CFR Part 50, Appendix B. The QA Program ensures that corrective actions are completed within the specific or general licensee's Corrective Action Program (CAP), and include provisions to: • Perform functionality assessments. • Perform apparent cause evaluations, and root cause evaluations. • Address the extent of condition. • Determine actions to prevent recurrence for significant conditions adverse to quality; ensure justifications for non-repairs. • Trend conditions. • Identify operating experience actions, including modification to the existing AMP (e.g., increased frequency). • Determine if the condition is reportable per 10 CFR 72.75. Example provisions are provided below, for assessment of extent of condition and evaluation criteria for canisters with aging effects, following from the example acceptance criteria. Alternative corrective actions to those provided below may be appropriate, per the licensees CAP evaluation to determine appropriate corrective actions to be taken if AMP acceptance criteria are not met. <u>Extent of Condition</u> Confirmation of localized corrosion or stress corrosion cracking may warrant inspection of additional canisters at the same ISFSI location to determine the extent of condition. Priority for additional inspections should be to canisters with similar time in service and initial loading. Canisters with confirmed localized corrosion or stress corrosion cracking must be evaluated for continued service. Canisters with localized corrosion or stress corrosion cracking that do not meet the prescribed evaluation criteria must be repaired or replaced.

Element	Description
7. Corrective Actions (cont'd)	<u>Evaluation Criteria for Canisters with Aging Effects</u> For austenitic stainless steel canisters covered by an AMP that utilizes the inspection and acceptance criteria in ASME B&PV code Section XI for Class 1 piping system, the disposition of canisters should be commensurate with in-service inspection results: • Canisters with no evidence of corrosion are permitted to remain in service and will continue to be evaluated at 5-year intervals. • Canisters with rust deposits that are determined to be a result of iron contamination but do not have evidence of localized corrosion or stress corrosion cracking are permitted to remain in service and will continue to be evaluated at 5-year intervals. • Canisters that show evidence of localized corrosion or stress corrosion cracking that does not exceed the acceptance standards in IWB-3514.1 are permitted to remain in service and will be evaluated at 5-year intervals. Sample size should be increased to assess canisters with similar susceptibility assessments. • Canisters that show evidence of localized corrosion or stress corrosion cracking that exceeds the acceptance standards in IWB-3514.1 but meet the acceptance criteria identified in IWB-3640 including the required evaluation per IWB-3641(a) using the prescribed evaluation procedures, are permitted to remain in service and should be evaluated at 3-year intervals. Sample size should be increased to assess canisters with similar susceptibility assessments. • Canisters that show evidence of localized corrosion or stress corrosion cracking that exceeds acceptance criteria identified in IWB-3640 are not permitted to remain in service without an engineering analysis and/or mitigation actions. Sample size should be increased to assess canisters with similar susceptibility assessments.
8. Confirmation Process	The confirmation process will be commensurate with the specific or general licensee Quality Assurance (QA) Program approved under 10 CFR Part 72, Subpart G or 10 CFR Part 50, Appendix B. The QA Program ensures that the confirmation process includes provisions to preclude repetition of significant conditions adverse to quality. The confirmation process describes or references procedures to: • Determine follow-up actions to verify effective implementation of corrective actions. • Monitor for adverse trends due to recurring or repetitive findings or observations.

Element	Description
9. Administrative Controls	The administrative controls are in accordance with the specific- or general-licensee QA Program approved under 10 CFR Part 72, Subpart G, or 10 CFR Part 50, Appendix B, respectively. The QA Program ensures that administrative controls include provisions that define: • instrument calibration and maintenance • inspector requirements • record retention requirements • document control The administrative controls describe or reference: • methods for reporting results to the NRC per 10 CFR 72.75 • frequency for updating AMP based on site-specific, design-specific, and industrywide operating experience

Element	Description
10. Operating Experience	The AMP references and evaluates applicable operating experience, both before renewal and will continue to do so as new operating experience is developed and made available after renewal, including: • internal and industrywide condition reports • internal and industrywide corrective action reports • vendor-issued safety bulletins • NRC Generic Communications • applicable DOE or industry initiatives (e.g., EPRI- or DOE-sponsored inspections) The AMP clearly identifies any degradation in the referenced operating experience as either age-related or event-driven, with proper justification for that assessment. Past operating experience supports the adequacy of the proposed AMP, including the method/technique, acceptance criteria, and frequency of inspection. The AMP references the methods for capturing operating experience from other ISFSIs with similar in-scope SSCs. CISCC of austenitic stainless steels is a known degradation mechanism for aqueous environments; however, operating experience in aqueous environments is not directly applicable in assessing the potential for atmospheric CISCC for austenitic stainless steel dry storage canisters. Atmospheric CISCC of austenitic stainless steels has been reported in a range of industries including welded stainless steel components and piping in operating nuclear power plants. <u>Spent Fuel Storage</u> Inspections of dry storage canisters after 20 years in service have been conducted at a few independent spent fuel installation (ISFSI) sites. Details of the inspection conducted at the Calvert Cliffs nuclear power plant ISFSI are documented in a recent EPRI report (Waldrop et al., 2014; Bryan and Enos, 2015). No evidence of localized corrosion was identified but some amount of chloride-containing salts were determined to be present and corrosion products believed to be related to iron contamination were identified. <u>Operating Power Reactors</u> NRC Information Notice 2012-20 (NRC, 2012) documents previous cases of atmospheric CISCC of welded stainless steel piping systems and tanks at operating reactor locations. Atmospheric CISCC growth rates determined from operating experience at both domestic and foreign nuclear power plants including events at San Onofre, Turkey Point, St. Lucie, and Koeberg (South Africa) range from 3.6×10^{-12} m/s to 2.9×10^{-11} m/s for components at ambient temperatures.

Element	Description
10. Operating Experience (cont'd)	Relevant Literature and Testing Electric Power Research Institute (EPRI) has recently conducted a literature review of CISCC which summarizes the results of many previous laboratory investigations (Gorman et al., 2014). The NRC has recently published the results of a completed investigation of CISCC testing of type 304, 304L and 316L stainless steel and welds (He, et al., 2014). This study indicates that SCC was initiated at stresses just above the yield strength in tests conducted using 304 stainless steel C-ring specimens. Testing with U-bend specimens showed that CISCC was observed with the lowest simulated sea salt concentrations tested (100 mg salt/m^2 or ~55 mg chloride/m^2) at temperatures of 52°C (125.6°F) using a maximum absolute humidity of 30 g/m^3, which is generally accepted as being near the maximum absolute humidity in a natural environment. Both laboratory and field investigations have been conducted by Central Research Institute of Electric Power Industry (CRIEPI) and Tokyo Electric Power Company (TEPCO). This includes the early work by Tokiwai et al. (1985) who reported the critical surface chloride concentrations of 8 mg/m^2 for CISCC on sensitized Type 304 stainless steel. Kosaka (2008) reported crack growth rates of 9.6×10^{-12} m/s obtained in natural exposure tests on Miyakojima Island with Type 304 base metals and welds, Type 304L welds and Type 316LN welds. Hayashibara, et al. (2008) reported activation energy for crack growth in Type 304 stainless steel of 5.6 to 9.4 kcal/mol (23 to 39 kJ/mol) based on testing conducted at temperatures of 50 to 80°C (122 to 176°F).

Element	Description
References	American Society of Mechanical Engineers, "Boiler and Pressure Vessel Code Section XI—Rules for Inservice Inspection of Nuclear Power Plant Components," New York, NY: ASME, 2007. Bryan, C.R., D.G. Enos, "Analysis of Dust Samples Collected from Spent Nuclear Fuel Interim Storage Containers at Hope Creek, Delaware, and Diablo Canyon, California," SAND2014-16383, Albuquerque, NM: Sandia National Laboratories, July 2014. Cumblidge, S.E., M.T. Anderson, S.R. Doctor, "An Assessment of Visual Testing," NUREG/CR 6860, Pacific Northwest National Laboratory, Richland, WA, 2004, ADAMS Accession No. ML043630040. Cumblidge, S.E., et al., "A Study of Remote Visual Methods to Detect Cracking in Reactor Components," NUREG/CR-6943, Pacific Northwest National Laboratory, Richland, WA, 2007, ADAMS Accession No. ML073110060. Fuhr, K., J. Broussard, G. White, "Susceptibility Assessment Criteria for Chloride-Induced Stress Corrosion Cracking (CISCC) of Welded Stainless Steel Canisters for Dry Cask Storage Systems," Electric Power Research Institute (EPRI), EPRI-3002005371, Palo Alto, CA, 2015. Gorman, J., K. Fuhr, J. Broussard, "Literature Review of Environmental Conditions and Chloride-Induced Degradation Relevant to Stainless Steel Canisters in Dry Cask Storage Systems," EPRI-3002002528, Palo Alto, CA, 2014. Hayashibara, H., et al. "Effect of Temperature and Humidity on Atmospheric Stress Corrosion Cracking of 304 Stainless Steel," NACE International, Corrosion 2008, paper 08492, Houston, TX, 2008. He, X., et al., "Assessment of Stress Corrosion Cracking Susceptibility for Austenitic Stainless Steels Exposed to Atmospheric Chloride and Non-Chloride Salts," NUREG/CR-7170, U.S. Nuclear Regulatory Commission, February 2014, ADAMS Accession No. ML14051A417. Kosaki, A., "Evaluation Method of Corrosion Lifetime of Conventional Stainless Steel Canister under Oceanic Air Environment," Nuclear Engineering and Design, Vol. 238, 2008, pp.1233–1240. Selby, G., "BWR Vessel and Internals Project, Reactor Pressure Vessel and Internals Examination Guidelines," EPRI 1011689, TR-105696-R8 (BWRVIP-03) Rev. 8, Palo Alto, CA, 2005. Tokiwai, M., H. Kimura, H. Kusanagi, "The Amount of Chlorine Contamination for Prevention of Stress Corrosion Cracking in Sensitized Type 304 Stainless Steel," Corrosion Science, Vol. 25 Issue 8–9, 1985, pp. 837–844.

Element	Description
References (cont'd)	U.S. Nuclear Regulatory Commission (NRC), Information Notice 2012-20: "Potential Chloride-Induced Stress Corrosion Cracking of Austenitic Stainless Steel and Maintenance of Dry Cask Storage System Canisters," November 14, 2012, ADAMS Accession No. ML12319A440. Waldrop, K. et al., "Calvert Cliffs Stainless Steel Dry Storage Canister Inspection," EPRI-1025209, EPRI, Palo Alto, CA, 2014.

Example AMP for Reinforced Concrete Structures

An example aging management program (AMP) for reinforced concrete structures is provided below. The AMP consists of condition monitoring, performance monitoring, mitigation and prevention activities. The program includes periodic visual inspections by personnel qualified to monitor reinforced concrete for applicable aging effects, such as those described in the American Concrete Institute (ACI) guides 349.3R-02, ACI 201.1R-08, and American National Standards Institute/American Society of Civil Engineers guidelines (ANSI/ASCE) 11-99. Identified aging effects are evaluated against acceptance criteria derived from the design bases or industry guides and standards, including ACI 349, ACI 318, ACI 349.3R-02 and the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section XI, Subsection IWL.

The program also includes periodic sampling and testing of groundwater and the need to assess the impact of any changes in its chemistry on below-grade concrete structures. Additional activities include radiation surveys to ensure the shielding functions of the concrete structure are maintained and daily inspections to ensure the air convection vents are not blocked, if applicable (per the requirements of the approved design bases). The program also includes provisions where modifications may be appropriate for specific license renewals.

Table B-2. Example AMP for Reinforced Concrete Structures

Element	Description
1. Scope of Program	The scope of the program includes the following aging management activities: 1. visual inspection of above-grade (accessible, inaccessible) and below-grade (underground) concrete areas (See Element 4 for sample size and justification of areas to be inspected) 2. groundwater chemistry monitoring program to identify conditions conducive to below-grade (underground) aging mechanisms: • corrosion of embedded steel • chemical attack (chloride and sulfate induced degradation 3. radiation surveys to: • Ensure compliance with 10 CFR 72.104 (i.e., dose equivalent requirements beyond the controlled area during normal and off-normal conditions of storage). • Monitor performance of the concrete as a neutron/gamma shield at near system locations as an indicator of concrete degradation. The program provides means to address the following aging effects and mechanisms, as described in ACI 349.3R-02 and ASCE/SEI 11-99: • cracking or loss of material (spalling, scaling) due to freeze-thaw degradation • cracking or loss of material (spalling, scaling) due to chemical attack (chloride, sulfate induced) • cracking and loss of strength due to cement aggregate reactions • cracking, loss of material, and loss of bond due to corrosion of embedded steel • increase in porosity/permeability and loss of strength due to leaching of calcium hydroxide ($Ca(OH)_2$) • cracking and distortion due to long-term settlement • cracking and reduction in strength due to gamma and neutron irradiation Calculations or analyses (time-limited aging analyses, when appropriate) may be used to demonstrate that aging effects due to irradiation do not require an AMP. More specifically, the renewal application may demonstrate that no part of the concrete exceeds critical cumulative fluences of 10^{17} neutrons/m^2 or 10^{10} rad (gamma dose), per ACI 349.3R-02.[a] Additional site-specific AMPs may be required for the following scenarios: • A dewatering system is used to prevent long-term settlement. • The design bases includes embedded aluminum subcomponents without a protective insulating coating. • Protective coatings are relied upon to manage the effects of aging for a subcomponent.

Element	Description
2. Preventive Actions	Preventive actions include (1) continuance of inspections to ensure that air inlet and outlet vents are not blocked, or (2) temperature monitoring, if applicable, to ensure design temperature limits are not exceeded. These inspections would be part of the approved design bases and be continued for the sample size and inspection frequency identified in the respective technical specification. Additional preventive actions are not required for structures designed and fabricated in accordance to ACI 318 or ACI 349, as specified in the design bases. Otherwise, a site-specific AMP may be required.
3. Parameters Monitored or Inspected	For visual inspections, the parameters monitored/inspected quantify the following aging effects: • cracking • material loss (spalling, scaling) • loss of bond • increased porosity/permeability The AMP references the following parameters for characterizing the above aging effects, as appropriate:[b] • affected surface area • geometry/depth of defect • cracking, crazing, delaminations, drummy areas • curling, settlements or deflections • honeycombing, bug holes • popouts and voids • exposure of embedded steel • staining/ evidence of corrosion • dusting, efflorescence of any color The parameters evaluated consider any surface geometries that may support water ponding and potentially increase the rate of degradation. For the groundwater chemistry program, the parameters monitored/inspected include: • water pH • concentration of chlorides and sulfates in the water For radiation surveys, the parameters monitored/inspected include gamma dose rate and/or neutron fluence rate.

Element	Description
4. Detection of Aging Effects	<u>Method/Technique</u> The method/technique achieves the acceptance criteria, as defined in AMP Element 6. An engineering justification or technical bases is provided, which references applicable consensus guides, codes and standards, or calibration procedures that ensure the method or technique will provide reliable data. For visual inspections, the method/technique is defined as: • visual method for normally accessible areas (e.g., feeler gauges, crack comparators) • visual method for normally inaccessible areas (site-qualified remote inspection system) Procedures for remote visual inspections should be demonstrated to ensure the acceptance criteria in ACI 349.3R is achievable; procedure attributes should include equipment resolution, lighting requirements, etc. and reference applicable standards when possible. For the groundwater chemistry program, the method/technique is defined as a chemical analysis method with a valid measurement range relative to the acceptance criteria. For radiation surveys, the method/technique is defined as calibrated neutron and gamma detectors with valid energy ranges. <u>Frequency of Inspection</u> The proposed inspection schedule is commensurate with ACI 349.3R-02. Alternative inspection frequencies provide a valid technical basis (engineering justification, operating experience data) for any deviation from ACI 349.3R-02. For visual inspections, the frequency of inspection is defined as: • for above-grade (accessible and inaccessible) areas: ≤ 5 years • for below-grade (underground) areas: ≤ 10 years, and when excavated for any reason • the use of opportunistic inspections in lieu of planned inspections per the above schedule provides a valid technical basis (engineering justification, operating experience data). For the groundwater chemistry program, the frequency of monitoring is justified (e.g., quarterly, semiannual). For radiation surveys, the frequency of monitoring is justified (e.g., quarterly).

<table>
<tr><th>Element</th><th>Description</th></tr>
<tr><td>4. Detection of Aging Effects (cont'd)</td><td><u>Sample Size</u>

Visual inspections cover 100 percent of readily accessible surfaces of all reinforced concrete structures in operation (e.g., all normally accessible exterior surfaces of all loaded overpacks), and 100 percent of normally inaccessible surfaces for a justified subset of the reinforced concrete structures in operation (e.g., interior surfaces of two overpacks, including the overpack earliest loaded and the overpack loaded with the highest heat-load canister). The extent of inspection coverage should be specified and demonstrated to sufficiently characterize the condition of the structure.

For the groundwater chemistry program and radiation surveys, the sample size clearly identifies and justifies specific locations where inspection/monitoring will be conducted to sufficiently characterize the condition of the structure (e.g., periodic dose rate measurements will be performed at same locations specified in the technical specifications for dose rate measurements at loading).

<u>Data Collection:</u>

Data collection for visual inspections is commensurate with applicable consensus codes/standards/guides:
• for example, ACI 224.1R for quantitative analysis (crack width, extent), ACI 562, ACI 364.1R.

The AMP references a clearinghouse for documenting inspection/monitoring operating experience.

<u>Timing</u>

The timing of the inspections includes the pre-application inspection or general-licensee baseline inspection, performed per Sections 3.4.1.2 and 3.6.1.10 of this NUREG, and at the frequency justified by the AMP.</td></tr>
<tr><td>5. Monitoring and Trending</td><td>Monitoring and trending methods are commensurate with defect evaluation guides and standards (e.g., ACI 201.1R, ACI 207.3R, ACI 364.1R, ACI 562, or ACI 224.1R for crack evaluation).

Monitoring and trending methods reference plans/procedures used to:
• Establish a baseline before or at the beginning of the period of extended operation.
• Track trending of parameters or effects not corrected in a previous inspection, for example:
  • crack growth/extent
  • pore/void density and affected areas
  • dose rates</td></tr>
</table>

Element	Description
6. Acceptance Criteria	For visual inspections, the acceptance criteria are commensurate with the three-tier quantitative criteria in ACI 349.3R-02: • acceptance without further evaluation • acceptance after review • acceptance requiring further evaluation The acceptance criteria clearly identify when a condition is to be entered in the Corrective Action Program (e.g., when Tier 2 acceptance per ACI 349.3R-02 is exceeded). For the groundwater chemistry program, the acceptance criteria are commensurate with ASME Code Section XI, Subsection IWL, which states that an aggressive below-grade environment is defined as: • pH < 5.5, chlorides > 500 ppm, or sulfates > 1500 ppm For radiation surveys, the acceptance criteria are justified and sufficient to ensure compliance with 10 CFR 72.104 and identify dose rates that statistically exceed calculated/expected dose rates at pre-determined measurement locations. The adequacy of the acceptance criteria considers measured dose rates versus calculated/expected dose rates for a dry storage system (DSS) given the DSS contents and accounting for the decay of the source term since the DSS loading. Measurement locations should be consistent with those specified in the license/CoC conditions/technical specification (if any) and/or locations where dose rates were calculated in the FSAR and likely measured at the time of loading. Alternative acceptance criteria may be provided. For such cases, the acceptance criteria should: • Include a quantitative basis (justifiable by operating experience, engineering analysis, consensus codes/standards). • Avoid use of non-quantifiable phrases (e.g., significant, moderate, minor, little, slight, few). • Be achievable and clearly actionable.
7. Corrective Actions	The corrective actions are in accordance with the specific- or general-licensee QA Program approved under 10 CFR Part 72, Subpart G, or 10 CFR Part 50, Appendix B, respectively. The QA Program ensures that corrective actions are completed within the specific or general licensee's CAP, and include provisions to: • Perform functionality assessments. • Perform apparent cause evaluations, and root cause evaluations. • Address the extent of condition. • Determine actions to prevent recurrence for significant conditions adverse to quality; ensure justifications for non-repairs. • Trend conditions. • Identify operating experience actions, including modification to the existing AMP (e.g., increased frequency). • Determine if the condition is reportable to the NRC per 10 CFR 72.75. The AMP references applicable concrete rehabilitation guides or standards, for example: • cracking: ACI 224.1R, ACI 562, ACI 364.1R, and ACI RAP Bulletins • spalling/scaling: ACI 562, ACI 364.1R, ACI 506R, and ACI RAP Bulletins

Element	Description
8. Confirmation Process	The confirmation process is commensurate with the specific- or general-licensee QA Program approved under 10 CFR Part 72, Subpart G, or 10 CFR Part 50, Appendix B, respectively. The QA Program ensures that the confirmation process includes provisions to preclude repetition of significant conditions adverse to quality. The confirmation process describes or references procedures to: • determine follow-up actions to verify effective implementation of corrective actions • monitor for adverse trends due to recurring or repetitive findings or observations
9. Administrative Controls	The administrative controls are in accordance with the specific- or general-licensee QA Program approved under 10 CFR Part 72, Subpart G, or 10 CFR Part 50, Appendix B, respectively. The QA Program ensures that the administrative controls include provisions that define: • instrument calibration and maintenance • inspector requirements (commensurate with ACI 349.3R-02) • record retention requirements • document control The administrative controls describe or reference: • methods for reporting results to the NRC per 10 CFR 72.75 • frequency for updating the AMP based on industrywide operating experience
10. Operating Experience	The AMP references and evaluates applicable operating experience, both before renewal and will continue to do so as new operating experience is developed and made available after renewal, including: • internal and industrywide condition reports • internal and industrywide corrective action reports • vendor-issued safety bulletins • NRC Generic Communications • applicable DOE or industry initiatives (e.g., EPRI- or DOE-sponsored inspections) The AMP clearly identifies any degradation in the referenced operating experience as either age-related or event-driven, with proper justification for that assessment. Past operating experience supports the adequacy of the proposed AMP, including the method/technique, acceptance criteria, and frequency of inspection. The AMP references the methods for capturing operating experience from other ISFSIs with similar in-scope SSCs.

Element	Description
References	ACI 201.1R-08, "Guide for Conducting a Visual Inspection of Concrete in Service," 2008. ACI 207.3R-94, "Practices for Evaluation of Concrete in Existing Massive Structures for Service Conditions," 2008. ACI 224.1R-07, "Causes, Evaluation, and Repair of Cracks in Concrete Structures," 2007. ACI 318-05, "Building Code Requirements for Structural Concrete and Commentary," 2005. ACI 349.3R-02, "Evaluation of Existing Nuclear Safety-Related Concrete Structures," 2010. ACI 349-06, "Code Requirements for Nuclear Safety-Related Concrete Structures and Commentary," 2007. ACI 364.1R-07, "Guide for Evaluation of Concrete Structures before Rehabilitation," 2007. ACI 506R-05, "Guide to Shotcrete," 2005. ACI 562-13, "Code Requirements for Evaluation, Repair, and Rehabilitation of Concrete Buildings," 2013. ACI CT-13, "ACI Concrete Terminology," 2013. ASCE/SEI 11-99, "Guideline for Structural Condition Assessment of Existing Buildings," 2000. ASME Boiler and Pressure Vessel Code, Section XI, Subsection IWL, "Requirements for Class CC Concrete Components of Light-Water-Cooled Plants," 2013. NRC, "Standard Review Plan for Spent Fuel Dry Storage Facilities," NUREG-1567, Rev. 0. Washington, DC, 2000. ADAMS Accession No. ML003686776. NRC, "Standard Review Plan for Spent Fuel Dry Storage Systems at a General License Facility," NUREG-1536, Rev. 1, Washington, DC, 2010a. ADAMS Accession No. ML101040620. NRC, "Generic Aging Lessons Learned (GALL) Report," NUREG-1801, Rev. 2, Washington, DC, 2010c. ADAMS Accession No. ML103490041.

[a] The staff recognizes the critical cumulative fluence in ACI 349.3R-02 as adequately conservative. The applicant may choose to justify a higher critical cumulative fluence if a technical bases is provided.
[b] Terminology consistent with ACI standard CT-13.

Example of a High Burnup Fuel Monitoring and Assessment Program

An example of a High Burnup (HBU) Fuel [1] Monitoring and Assessment Program is provided below. This is a licensee program that monitors and assesses data and other information regarding HBU fuel performance, to confirm that the design-bases HBU fuel configuration is maintained during the period of extended operation. This example HBU Monitoring and Assessment Program relies on a surrogate demonstration program to provide data on HBU fuel performance. Guidance for determining if a surrogate demonstration program can provide the data to support a licensee's HBU Fuel Monitoring and Assessment Program is given in Appendix D. Although this example focuses on the use of a surrogate demonstration program, a licensee may use alternative approaches that are appropriately justified.

The aging management review is not expected to identify any aging effects that could lead to fuel reconfiguration, as long as the HBU fuel is stored in a dry inert environment, temperature limits are maintained, and thermal cycling is limited (see Sections 2.4.2.1 and 3.4.1.4). Short-term testing (i.e., laboratory scale testing up to a few months) and scientific analyses examining the performance of HBU fuel have provided a foundation for the technical basis that storage of HBU fuel in the period of extended operation may be performed safely and in compliance with regulations. However, there has been relatively little operating experience, to date, with dry storage of HBU fuel.

Therefore, the purpose of a HBU Fuel Monitoring and Assessment Program is to monitor and assess data and other information regarding HBU fuel performance to confirm there is no degradation of HBU fuel that would result in an unanalyzed configuration during the period of extended operation. The following description of an example HBU Fuel Monitoring and Assessment Program presents the applicable information in a format using each element of an effective aging management program, to provide a framework for such a monitoring and assessment program.

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[1] Fuel assemblies with discharge burnup greater than 45 gigawatt-days per metric ton of uranium (GWd/MTU)

Table B-3. Example of a High Burnup Fuel Monitoring and Assessment Program

Element	Description
1. Scope of the Program	The scope of the program provides a description of: (1) the design-bases characteristics of the HBU fuel, (2) the surrogate demonstration program that will be used to provide data on the applicable design-bases HBU fuel performance, and (3) how the parameters of the surrogate demonstration program are applicable to the design-bases HBU fuel. Aging effects will be determined for material/environment combinations through an alternative surrogate demonstration program meeting the guidance in Appendix D. Example language to address this "scope of the program" element follows: Fuel stored in a *[define cask/canister model]* is limited to an assembly average burnup of *[define design-bases limit]* GWd/MTU. The cladding materials for the HBU fuel are *[define types of cladding],* and the fuel is stored in a dry helium environment. HBU fuel was first placed into dry storage in a *[define cask/canister model]* on *[start date of storage term of first storage of HBU fuel].* The program relies on the joint Electric Power Research Institute (EPRI) and Department of Energy (DOE) "HBU Dry Storage Cask Research and Development Project" (HDRP) (EPRI 2014), conducted in accordance with the guidance in Appendix D, as a surrogate demonstration program that monitors the performance of HBU fuel in dry storage. The HDRP is a program designed to collect data from a spent nuclear fuel storage system containing HBU fuel in a dry helium environment. The program entails loading and storing a AREVA TN-32 bolted lid cask (the "Research Project Cask") at Dominion Virginia Power's North Anna Power Station with intact HBU fuel (of average assembly burnups ranging between 53 GWd/MTU and 55.5 GWd/MTU). The fuel to be used in the program include four kinds of cladding (Zircaloy-4, low-tin Zircaloy-4, Zirlo^{TM}, and M5^{TM}). The Research Project Cask is to be licensed to the temperature limits contained in Interim Staff Guidance (ISG) 11, Rev. 3 (NRC 2003), and loaded such that the fuel cladding temperature is as close to the limit as practicable. *[If an alternative surrogate demonstration program is used, provide a description of the program.]*

Element	Description
1. Scope of the Program (cont'd)	The parameters of the surrogate demonstration program are applicable to the design-bases HBU fuel, as the: (1) maximum assembly-average burnup of the design-bases HBU fuel *[define value]* is less than the assembly-average burnup of the fuel in the surrogate demonstration program *[define value]*; (2) the cladding type of the design-bases HBU fuel *[define type]* is the same as the surrogate demonstration program *[define type]*; and (3) the peak temperatures in the surrogate demonstration program *[define values]* bound the peak temperatures of the loaded systems (or design bases) *[define values]*.[a]
2. Preventive Actions	There are no specific preventive actions associated with this HBU Fuel Monitoring and Assessment Program. However, the applicant should discuss the design-bases characteristics of the licensed/certified dry storage system, in terms of initial cask loading operations, to show the HBU fuel is stored in a dry inert environment. Example language follows: During the initial loading operations of the cask/canister, the design and ISFSI Technical specifications (TS) require that the fuel be stored in a dry inert environment. TS *[name and number]* demonstrates that the cask/canister cavity is dry by maintaining a cavity absolute pressure less than or equal to *[value]* for a *[time period]* with the cask/canister isolated from the vacuum pump. TS *[name and number],* requires that the cask/canister then be backfilled with helium. These two TS requirements ensure that the HBU fuel is stored in an inert environment thus preventing cladding degradation due to oxidation mechanisms. TS *[name and number]* also requires that the helium environment be established within *[time]* hours of commencing cask/canister draining. The cask/canister is loaded in accordance with the criteria of ISG-11.
3. Parameters Monitored or Inspected	The applicant identifies the parameters monitored and inspected in a surrogate demonstration program that are applicable to its particular design-bases HBU fuel and describes how this meets the guidance of Appendix D.
4. Detection of Aging Effects	The applicant identifies the detection of aging effects in a surrogate demonstration program that are applicable to its particular design-bases HBU fuel and describes how this meets the guidance of Appendix D.

Element	Description
5. Monitoring and Trending	As information/data from a surrogate demonstration program or from other sources (such as testing or research results and scientific analyses) becomes available, the licensee will monitor, evaluate, and trend the information via its operating experience program and/or the Corrective Action Program (CAP) to determine what actions should be taken. The licensee will evaluate the information/data from a surrogate demonstration program or from other sources to determine whether the acceptance criteria in Element 6 are met. • If all of the acceptance criteria are met, no further assessment is needed. • If any of the acceptance criteria are not met, the licensee must conduct additional assessments and implement appropriate corrective actions (see Element 7). Formal evaluations of the aggregate information from a surrogate demonstration program and other available domestic or international operating experience (including data from monitoring and inspection programs, NRC generic communications, and other information) will be performed at specific points in time during the period of extended operation, as delineated in Table B-4.
6. Acceptance Criteria	The High Burnup Fuel Monitoring and Assessment Program acceptance criteria are: • Hydrogen content—maximum hydrogen content of the cover gas over the approved storage period should be extrapolated from the gas measurements to be less than the design-bases limit for hydrogen content. • Moisture content—the moisture content in the cask/canister, accounting for measurement uncertainty, should be less than the expected upper bound moisture content per the design-bases drying process.[b] • Fuel condition/performance—nondestructive examination (e.g., fission gas analysis) and destructive examination (e.g., to obtain data on creep, fission gas release, hydride reorientation, cladding oxidation, and cladding mechanical properties) should confirm the design-bases fuel condition (i.e., no changes to the analyzed fuel configuration considered in the safety analyses of the approved design bases). The applicant should provide information on the design-bases characteristics of the dry storage system or ISFSI, with regard to these criteria. The applicant should reference the source of specific values, or explain any assumptions made, for defining design-bases characteristics of the fuel condition/performance.

Element	Description
7. Corrective Actions	The corrective actions are in accordance with the specific- or general-licensee QA Program approved under 10 CFR Part 72, Subpart G, or 10 CFR Part 50, Appendix B, respectively. Corrective actions should be implemented if data from a surrogate demonstration program or other sources of information indicate that any of the High Burnup Fuel Monitoring and Assessment Program acceptance criteria (in Element 6) are not met. If any of the acceptance criteria are not met, the licensee will: (1) Assess fuel performance (effects on fuel and changes to fuel configuration). (2) Assess the design-bases safety analyses, considering degraded fuel performance (and any changes to fuel configuration), to determine the ability of the dry storage system/ISFSI to continue to perform its intended functions under normal, off-normal, and accident conditions. The licensee will determine what corrective actions should be taken to: (1) Manage fuel performance, if any. (2) Manage impacts related to degraded fuel performance to ensure that all intended functions for the dry storage system/ISFSI are met. In addition, the licensee will obtain the necessary NRC approval in the appropriate licensing/certification process for modification of the design bases to address any conditions outside of the approved design bases.
8. Confirmation Process	The confirmation process is commensurate with the specific- or general-licensee QA Program approved under 10 CFR Part 72, Subpart G, or 10 CFR Part 50, Appendix B, respectively. The QA Program ensures that the confirmation process includes provisions to ensure corrective actions are adequate and appropriate, have been completed, and are effective. The focus of the confirmation process is on the follow-up actions that must be taken to verify effective implementation of corrective actions. The measure of effectiveness is in terms of correcting the adverse condition and precluding repetition of significant conditions adverse to quality. Procedures include provisions for timely evaluation of adverse conditions and implementation of any corrective actions required, including root cause evaluations and prevention of recurrence where appropriate. These procedures provide for tracking, coordinating, monitoring, reviewing, verifying, validating, and approving corrective actions, to ensure effective corrective actions are taken.
9. Administrative Controls	The administrative controls are in accordance with the specific- or general-licensee QA Program approved under 10 CFR Part 72, Subpart G, or 10 CFR Part 50, Appendix B, respectively. The QA Program ensures that the administrative controls include provisions that define: • formal review and approval processes • record retention requirements • document control

Element	Description
10. Operating Experience	The program references and evaluates applicable operating experience, both before renewal and will continue to do so as new operating experience is developed and made available after renewal, including: • internal and industrywide condition reports • internal and industrywide corrective action reports • vendor-issued safety bulletins • NRC Generic Communications • applicable DOE or industry initiatives (e.g., HDRP) • applicable research (e.g., Oak Ridge National Laboratory studies on bending responses of the fuel, Argonne National Laboratory and Central Research Institute of Electric Power Industry studies on hydride reorientation effects) The review of operating experience clearly identifies any HBU fuel degradation as either age-related or event-driven, with proper justification for that assessment. Past operating experience supports the adequacy of the HBU Fuel Monitoring and Assessment Program. Surrogate demonstration programs with storage conditions and fuel types similar to those in the licensed ISFSI/certified dry storage system that meet the guidance in Appendix D are a viable method to obtain operating experience. New data/research on fuel performance from both domestic and international sources that are relevant to the licensed/certified HBU fuel in the dry storage system/ISFSI should be evaluated on a periodic basis and the AMP updated and revised as needed.

Element	Description
References	Center for Nuclear Waste Regulatory Analyses, "Extended Storage and Transportation: Evaluation of Drying Adequacy," June 2013, NRC Contract No. NRC-02-07-006, 2013. ADAMS Accession No. ML13169A039. Dominion Resources, Inc., "North Anna ISFSI, TN-32 High Burnup Dry Storage Research Project," May 13, 2015. ADAMS Accession No. ML15133A082. EPRI, "HBU Dry Storage Cask Research and Development Project, Final Test Plan," February 27, 2014, DOE Contract No. DE-NE-0000593. NRC, "Standard Review Plan for Spent Fuel Dry Storage Systems at a General License Facility," NUREG-1536, Rev. 1, Washington, DC, 2010. ADAMS Accession No. ML101040620. NRC, Interim Staff Guidance 11, "Cladding Considerations for the Transportation and Storage of Spent Fuel," Revision 3, November 17, 2003. ADAMS Accession No. ML033230335.

[a] The AMP should provide a technical basis that supports the conclusion that the demonstration fuel is reasonably characteristic of the licensee's stored fuel, in the event of differences in burnup, cladding composition (alloy types) and peak cladding temperatures observed during loading (See Appendix D).
[b] The applicant will need to provide the expected upper bound moisture content based on its design-bases drying process. For example, if the design-bases drying process involves a vacuum drying method of evacuating a cask/canister to less than or equal to 3 torr and maintaining a constant pressure for 30 minutes after the cask/canister is isolated from the vacuum pump, the expected water content is about 0.43 gram-mole. (See NUREG-1536, Rev. 1.)

Table B-4. Formal Evaluations of Aggregate Information on HBU Fuel Performance

	Year	Assessment
1	Date—before HBU fuel exceeds the initial storage term [a]	Evaluate information obtained from a surrogate demonstration program for loading and initial period of storage along with other available sources of information. If surrogate demonstration program nondestructive examination (NDE) (i.e., cask/canister gas sampling, temperature data) data has not been obtained at this point, then the licensee has to provide evidence that the acceptance criteria in Element 6 are met or initiate a corrective action.
2	Date—10 years after Assessment 1 above	Evaluate, if available, information obtained from the destructive examination and NDE of the fuel placed into storage in a surrogate demonstration program along with other available sources of information. If the destructive examination data from a surrogate demonstration program has not been obtained at this point, then the licensee has to provide evidence to the NRC, by opening a cask/canister or separate effects surrogate experiments, that the acceptance criteria in Element 6 are met or initiate a corrective action.
3	Date—10 years after Assessment 2 above	Evaluate any new information.

[a] See Appendix F for a discussion of storage terms.

APPENDIX C

[RESERVED]

Upon consideration of public comments received on Draft NUREG-1927, Revision 1, the concepts regarding lead system inspections originally presented in Appendix C of the Draft NUREG-1927, Revision 1, have been incorporated into Chapter 3, "Aging Management Review," of this Final NUREG-1927, Revision 1. Therefore, the content in Appendix C has been deleted, and Appendix C is reserved for future use.

APPENDIX D

SUPPLEMENTAL GUIDANCE FOR THE USE OF A DEMONSTRATION PROGRAM AS A SURVEILLANCE TOOL FOR CONFIRMATION OF INTEGRITY OF HIGH BURNUP FUEL DURING THE PERIOD OF EXTENDED OPERATION

Appendix D

Supplemental Guidance for the Use of a Demonstration Program as a Surveillance Tool for Confirmation of Integrity of High Burnup Fuel During the Period of Extended Operation

This guidance provides the U.S. Nuclear Regulatory Commission (NRC) staff a basis for reviewing if a demonstration of high burnup (HBU) fuel has the necessary properties to qualify as one method that an applicant might use in license and certificate of compliance renewal applications to confirm the integrity of HBU fuel during continued storage.

D.1 Discussion

The experimental confirmatory basis that low burnup fuel (≤45 GWd/MTU) will maintain its integrity in dry cask storage over extended time periods was provided in a demonstration test (NRC, 2003; Bare, et al., 2001; Einziger, et al., 2003). A similar confirmation test, which includes information over a similar length of the time available for low burnup fuel, does not exist for other light water reactor (LWR) fuels, HBU fuel [1] and mixed oxide fuels. Certification and licensing HBU fuel for storage was permitted for an initial 20-year-term using the guidance contained in Interim Staff Guidance (ISG) 11 (NRC 2003), which was based on short-term laboratory tests and analysis that may not be applicable to the storage of HBU fuel beyond 20 years, particularly with the current state of knowledge regarding HBU fuel cladding properties.

One concern stated in ISG-11 was the potential detrimental effects, such as reduced ductility, of hydride reorientation on cladding behavior (NRC, 2003). Research performed in Japan and the United States indicated that: (1) hydrides could reorient at a significantly lower stress than previously believed (Billone, et al., 2013; Kamimura, 2010; Daum, et al, 2006), and (2) the radial hydrides could raise the cladding ductile-to-brittle transition temperature enough to compromise the ability of the cladding to withstand stress without undergoing brittle failure (Billone, et al., 2013). This phenomenon could influence the retrievability of HBU fuel assemblies and result in operational safety concerns in the handling of individual assemblies as HBU fuel cooled. Circumferential zirconium hydrides in the fuel cladding regions would dissolve into the fuel cladding during drying and reprecipitate (reorient) as radial hydrides as the fuel cladding cooled. Thus, fuel cladding with radial hydrides that is below a ductile-to-brittle transition temperature could be too brittle to retrieve (remove from the DSS) on an assembly basis. The maximum temperatures and internal rod pressures in ISG-11 were recommended to mitigate hydride reorientation and are applicable to HBU fuel during the initial 20-year storage, as the decay heat of HBU fuel is expected to maintain cladding temperatures above a ductile-to-brittle transition temperature (about 200 degrees Celsius, or about 392 degrees Fahrenheit).

There is no evidence to suggest that HBU fuel cannot similarly be stored safely and then retrieved for time periods beyond 20 years, but the supporting experimental data is not extensive. Therefore, confirmatory data or a commitment to obtain data on HBU fuel and taking appropriate steps in an aging management plan (AMP) will provide further information that will be useful in evaluating the safe handling of individual assemblies of HBU fuel for extended durations.

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[1] High burnup fuel is fuel with burnup ≥ 45 GWd/MTU.

A demonstration program could provide an acceptable method for an applicant to demonstrate compliance with the cited regulations for storage of light water reactor fuels (LWR) for periods of greater than 20 years by:

- confirming the expected fuel conditions, based on technical arguments made in ISG-11 (NRC, 2003), after a substantial storage period that is sufficiently long (about 10 years) to extrapolate the findings to the storage duration of interest (The behavior of the cladding for the period of extended operation will depend on its physical condition at the end of the initial 20-year storage period.)
- providing data for benchmarking, confirming predictive models, and updating aging management plans
- confirming the time-limited aging analysis (TLAA) cladding creep predictions that are the basis for the guidance recommendation for the maximum temperature in ISG-11 (NRC, 2003) are not exceeded and that sufficient creep margin exists for the extended storage period
- determining the system is sufficiently dry to eliminate moisture-driven degradation from consideration
- providing operating experience on the fuel behavior and drying procedure as input to an AMP on the behavior of the fuel
- identifying any aging effects that may be missed through short-term accelerated studies and analyses

Monitoring of the fuel temperatures, gas composition, and other conditions in the canister or cask combined with physical examination of the fuel at periodic intervals should be able to provide confirmation if:

- The models of the phenomena used for the first 20-year predictions can be used for the TLAA beyond 20 years.
- The condition of the fuel, after an appropriately long period of storage, does not degrade.
- New degradation mechanisms are not being exhibited.

Extrapolation outside the recorded data carries risk, but that risk can be minimized if the length of the extrapolation is reduced and those extrapolations are updated as the demonstration continues to monitor and measure fuel properties.

D.2 Technical Review Guidance

The applicant may use the results of a completed demonstration or an ongoing demonstration if the conditions of the demonstration meet the requirements stated below for the fuels and conditions of storage for which the term is to be renewed. The approach in this guidance can be applied to a generic demonstration program or a site/system-specific program as long as the demonstration's parameters are reasonably applicable to the applicant's fuel type and characteristics.

The technical reviewers should establish that the following conditions are met if the demonstration is to be used by the applicant to support fuel assembly conditions for storage of HBU fuel beyond 20 years and to be applicable to support a license or certificate application:

(1) That the maximum burnup of the fuel in the application is less than the burnup of the fuel in the demonstration. If the burnup is higher than that in the demonstration, the applicant should provide evidence, based on characteristics of the fuel, derived either from reactor rod qualification testing or other separate effects tests, that the demonstration fuel is reasonably characteristic of the stored fuel and the added burnup will not change the results determined by the demonstration. Similarly, if there is a different cladding type used, arguments based on comparison of composition and fabrication technique (e.g., stress-relieved and annealed, recrystallized) should justify the use of the demonstration results.

(2) If the applicant uses direct observations of the rod behavior to imply the condition of the rods in its system, either (1) the temperatures in the demonstration must bound the temperatures in the application, or (2) if the applicant uses predictive tools that have been confirmed by the demonstration, then the temperatures of the rods in the application do not have to be bounded by the temperature of the rods in the demonstration. The temperature models used in the application should either be benchmarked (1) against the demonstration temperature data, or (2) against actual measured rod temperature data in the same temperature range.

(3) If the applicant is using gas analysis or another gas detection method to establish the condition of the fuel, then the interior of a demonstration canister or cask should be quantitatively monitored for, at a minimum, moisture, oxygen, and fission gas. The duration and frequency of the gas monitoring should be determined by analysis of the potential degradation. Gases should always be quantitatively monitored before opening of the canister. If the applicant claims that no galvanic degradation is feasible, then, if after drying, moisture is detected in the canister, moisture and hydrogen should be monitored at a reasonable frequency to be determined by the applicant until the moisture disappears. Gas monitoring is not expected during movement of the canister. If the applicant is using the gas analysis to show no breaches would occur during transport, gas quantitative monitoring must be conducted before and after transport.

(4) Temperature monitoring should be conducted at a frequency that is suitable for determining the profile over the duration of the demonstration.

(5) If possible, some population of stored rods should be examined whenever the system is opened. These rods should be extracted from the fuel assembly to determine properties of the rods that affect degradation such as cladding creep, fission gas release, hydride reorientation, cladding oxidation, and mechanical properties.

(6) The demonstration program fuel shall include at least two full fuel assemblies. The assemblies may be reconstituted.

(7) Data from the demonstration program must be indicative of a storage duration long enough to justify extrapolation to the total storage time requested but no less than

10 years if the data is to be used to support license extension from the initial 20 years to an additional 40 years.[2]

D.3 References

Bare, W.C, L.D. Torgerson, "Dry Cask Storage Characterization Project—Phase 1: CASTOR V/21 Cask Opening and Examination," NUREG/CR-6745, Idaho National Engineering and Environmental Laboratory, Idaho Falls, ID, 2001. Agencywide Documents Access and Management System (ADAMS) Accession No. ML013020363.

Billone, M.C., T.A. Burtseva, R.E. Einziger, "Ductile-to-Brittle Transition Temperature for High-Burnup Cladding Alloys Exposed to Simulated Drying-Storage Conditions," Journal of Nuclear Materials, Volume 433, Issues 1–3, pages 431–448, February 2013.

Daum, R.S., et al., 2006. "Radial-Hydride Embrittlement of High-burnup Zircaloy-4 Cladding", Journal of Nuclear Science and Technology, Vol. 43, No. 9, p.1054, 2006.

Einziger, R.E., et al., "Examination of Spent PWR Fuel Rods after 15 Years in Dry Storage," NUREG/CR-6831, Argonne National Laboratory, Argonne, IL, 2003. ADAMS Accession No. ML032731021.

Kamimura, K. "Integrity Criteria of Spent Fuel for Dry Storage in Japan," Proceedings of the International Seminar on Interim Storage of Spent Fuel, ISSF 2010, page VI3-1, Tokyo November 2010.

U.S. Nuclear Regulatory Commission (NRC), Interim Staff Guidance 11, "Cladding Considerations for the Transportation and Storage of Spent Fuel," Rev. 3, November 17, 2003.

NRC, "The Use of a Demonstration Program as a Surveillance Tool for Confirmation of Integrity for Continued Storage of HBU Fuel Beyond 20 Years," ISG-24, Revision 0. Washington, DC, 2014. ADAMS Accession No. ML14058B166.

[2] A demonstration is to provide that there was satisfactory performance during the first 20 years and that the results could be extrapolated to support an additional 40 years. The NRC staff agreed that a demonstration of ≤ 10 years storage duration is insufficient to support these goals.

APPENDIX E

CONSIDERATIONS FOR RENEWALS OF CERTIFICATES OF COMPLIANCE

Appendix E

Considerations for Renewals of Certificates of Compliance

E.1 Development of Time-Limited Aging Analyses and Aging Management Programs

From Title 10 of the *Code of Federal Regulations* (10 CFR) 72.240, "Conditions for Spent Fuel Storage Cask Renewal," the certificate of compliance (CoC) renewal application must include time-limited aging analyses (TLAAs), if applicable, and aging management programs (AMPs). The CoC holder, as the applicant for the CoC renewal and the owner of the storage cask or dry storage system (DSS) design, will develop the TLAAs and AMPs. The renewal application should address age-related degradation of the DSS design in a bounding manner (i.e., fully address pertinent aging mechanisms and effects in all possible service environments where the DSS is being used and can be used). If there is an AMP that may not be applicable to certain general licensees, because of the service environment in which the DSS is located, the AMP should specify this. In addition, the CoC holder and the staff should consider the need for a CoC condition to specify this potential limited use of the AMP, so it is clear for general licensee implementation.

For a CoC renewal that encompasses CoC amendments with different design bases, the CoC holder will need to address in the renewal application how the TLAAs or AMPs apply to each amendment covered by the CoC. For example, if different materials are used or different SSCs are part of the DSS design in different CoC amendments, or if different environments (e.g., underground vs. aboveground system) are reflected in different CoC amendments, then there may be different TLAAs or AMPs specified for the individual amendments.

E.2 Implementation of AMPs

In approving the renewal of the DSS design, the U.S. Nuclear Regulatory Commission (NRC) may revise the CoC to include terms, conditions, and specifications that will ensure the safe operation of the DSS during the period of extended operation, including but not limited to, terms, conditions, and specifications that will require the implementation of an AMP by a general licensee, in accordance with 10 CFR 72.240(e). General licensees' implementation of AMPs is subject to NRC inspection.

Regulations in 10 CFR 72.212, "Conditions of General License Issued under § 72.210," provide requirements for general licensees using approved CoCs. Regulations in 10 CFR 72.212(b)(11) require general licensees to comply with the terms, conditions, and specifications of the CoC, including but not limited to, the requirements of any AMP put into effect as a condition of the NRC approval of a CoC renewal application in accordance with 10 CFR 72.240.

General licensees (CoC users) are responsible for implementing the AMPs. To document the licensee's compliance with the renewed CoC, a future or existing general licensee should include in its 10 CFR 72.212(b)(5) evaluation (including the results of the review and determination per 72.212(b)(6) and 72.212(b)(8)) how it will meet the new CoC terms, conditions, or specifications for aging management. Note that the renewed CoC may include a condition for the general licensee to do so. As part of this evaluation, general licensees should consider any granted exemptions or 10 CFR 72.48 changes that could affect aging management.

The general licensee should update the 10 CFR 72.212(b)(5) evaluation before entering the period of extended operation. Considering timely renewal provisions (See Section 1.4.5), update of the 10 CFR 72.212(b)(5) evaluation before the loaded systems enter the period of extended operation may not be possible. In such cases, the reviewer should ensure that timing for update of the 10 CFR 72.212(b)(5) evaluation is addressed in the application in a clear manner. Also, any CoC condition related to the update of the 10 CFR 72.212(b)(5) evaluation may include timing provisions.

The AMP-related information in the DSS FSAR will be implemented by general licensees as-written. If a general licensee wishes to deviate from the DSS FSAR, it must evaluate any such deviation under the provisions of 10 CFR 72.48. If AMP details in the FSAR specify that the AMP is not applicable for certain users (e.g., if it is not applicable in certain climates or environments), the general licensee can include the technical justification in its 10 CFR 72.212 report for not implementing such an AMP. The general licensee would need to evaluate any changes to the 10 CFR 72.212 report or any deviations from the DSS FSAR, using the requirements of 10 CFR 72.48 (See NRC Regulatory Issue Summary 2012-05, "Clarifying the Relationship Between 10 CFR 72.212 and 10 CFR 72.48 Evaluations").

E.3 Corrective Actions

As discussed in Section 3.6.1.7, corrective actions are measures to be taken when the AMP acceptance criteria are not met. Corrective actions are critical for maintaining the intended functions of the structures, systems, and components during the initial storage term as well as the period of extended operation. The CoC holder should discuss in its renewal application any applicable and appropriate corrective actions to be taken if the AMP acceptance criteria are not met.

A general licensee will use its Corrective Action Program (CAP) (that is consistent with the criteria in 10 CFR Part 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants") to capture and address aging effects identified in the period of extended operation. At a minimum, all conditions that do not meet the AMP acceptance criteria should be entered into the CAP. The general licensee's CAP should be able to respond to and adequately address and rectify any ISFSI or DSS aging issues. Also, the CAP's response to address any DSS aging issues should include any specific corrective actions specified in the CoC renewal application.

APPENDIX F

STORAGE TERMS

Appendix F

Storage Terms

F.1 Introduction

This appendix provides a flow chart for calculating storage terms of a dry storage system (DSS) or a DSS structure, system, and component (SSC) loaded during either the initial storage period or renewal period(s) of a certificate of compliance (CoC).

F.2 Storage Term Defined

The storage term (length of time a DSS can remain loaded) is determined by the period specified in the applicable CoC in effect at the time the DSS is placed into service (from Title 10 of the *Code of Federal Regulations* (10 CFR) 72.212(a)(3) and Ref. 1). The storage period begins when the DSS is first used by the general licensee to store spent fuel (10 CFR 72.212(a)(3)).[1] The clock starts when the loaded cask has been deployed in the independent spent fuel storage installation (76 FR 8872). If a CoC is not renewed, upon expiration, casks loaded under that CoC would need to be removed from service when they reach the end of their storage term. An AMP for a renewed CoC commences at the end of the initial storage period for each loaded DSS (see Section 3.6.2).

- If the DSS is loaded during the initial CoC term (e.g., 20 years) and the CoC is not renewed, the storage term is the entirety of the initial CoC term (e.g., 20 years).
- If the DSS is loaded during the initial CoC term and the CoC is renewed once, the storage term is the remaining time in the initial CoC term added to the entirety of the first renewal period.
- If the DSS is loaded during the first renewal period (e.g., 40 years), and the CoC is not subsequently renewed, the storage term is the entirety of the first renewal period (e.g., 40 years).
- If the DSS is loaded during the first renewal period, and the CoC is subsequently renewed, the storage term is the remaining time in the first renewal period added to the entirety of the subsequent renewal period (cumulative).

F.3 Flowchart for Calculating Storage Terms

A flowchart is provided below to assist the user in calculating the storage term for a DSS loaded under a CoC.

[1] The storage period for a particular DSS SSC begins when it is first used to store spent fuel, regardless if the SSC is later stored at a different location.

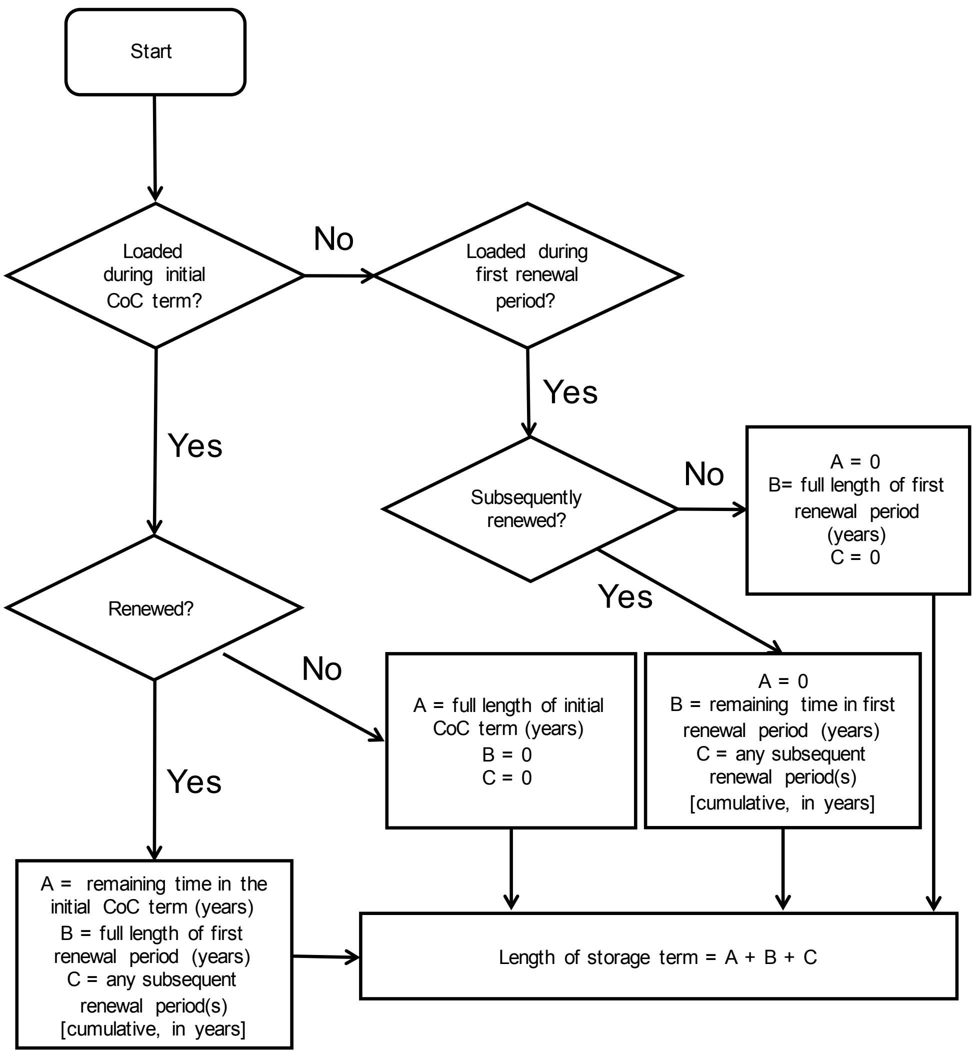

Figure F-1. Flowchart for calculating storage terms

F.4 References

U.S. Nuclear Regulatory Commission, “10 CFR Part 72, License and Certificate of Compliance Terms, Final Rule,” *Federal Register*, Vol. 76, No. 32, February 16, 2011, pp. 8872–8892 (76 FR 8872).

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10. SUPPLEMENTARY NOTES

11. ABSTRACT (200 words or less)

This Standard Review Plan is intended for use by U.S. Nuclear Regulatory Commission (NRC) reviewers. It provides guidance for the safety review of renewal applications for specific licenses of independent spent fuel storage installations and certificates of compliance (CoCs) of dry storage systems, as codified in Title 10 of the Code of Federal Regulations (10 CFR) Part 72, "Licensing Requirements for the Independent Storage of Spent Nuclear Fuel, High-Level Radioactive Waste, and Reactor-Related Greater Than Class C Waste."

To renew a specific license, an applicant (i.e., licensee) must submit a license renewal application at least 2 years before the expiration of the license in accordance with the requirements of 10 CFR 72.42(b). To renew a CoC, an applicant (i.e., CoC holder, user, or user's representative) must submit a renewal application at least 30 days before the expiration of the associated CoC in accordance with the requirements of 10 CFR 72.240(b). The NRC may renew a specific license or a CoC for a term not to exceed 40 years, in accordance with 10 CFR 72.42(a), or 10 CFR 72.240(a), respectively.

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NUREG-1927, Rev. 1 Final	**Standard Review Plan for Renewal of Specific Licenses and Certificates of Compliance for Dry Storage of Spent Nuclear Fuel**	**June 2016**

Division of Spent Fuel Storage and Transportation
Interim Staff Guidance - 1, Revision 2
Classifying the Condition of Spent Nuclear Fuel for Interim Storage and Transportation Based on Function[1]

Issue

This Interim Staff Guidance (ISG) provides guidance to the staff on classifying spent nuclear fuel as either (1) damaged, (2) undamaged, or (3) intact, before interim storage or transportation. This is not a regulation or requirement and can be modified or superseded by an applicant with supportable technical arguments.

Regulatory Basis

Fuel-Specific Regulations:

A fuel-specific regulation means a characteristic or performance requirement of the fuel specifically named in the applicable Code of Federal Regulations (CFR). These are regulations that specify capabilities that the spent nuclear fuel (SNF) must have. Examples include:

10 CFR 71.55(d) states, in part: "A package used for the shipment of fissile material must be so designed and constructed and its contents so limited that under the tests specified in 10 CFR 71.71 ('Normal conditions of transport')

(1) The contents would be subcritical.
(2) The geometric form of the package contents would not be substantially altered."

10 CFR 72.44(c) states, in part: "Technical specifications must include requirements in the following categories: (1) Functional and operating limits . . . (I) . . . for an ISFSI or MRS are limits on fuel or waste handling and storage conditions that are found to be necessary to protect the integrity of the stored fuel or waste container, to protect employees against occupational exposures and to guard against the uncontrolled release of radioactive materials . . . "

10 CFR 72.122(h)(1) states: "The spent fuel cladding must be protected during storage against degradation that leads to gross ruptures or the fuel must be otherwise confined such that degradation of the fuel during storage will not pose operational safety problems with respect to its removal from storage. This may be accomplished by canning of consolidated fuel rods or unconsolidated assemblies or other means as appropriate."

10 CFR 72.122(l) states: "Retrievability. Storage systems must be designed to allow ready retrieval of spent fuel, . . . for further processing or disposal."

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[1]Formerly entitled "Damaged Fuel."

System-Related Regulations:

A transportation and storage system must satisfy all applicable regulations in 10 CFR Parts 71 or 72. A system-related regulation is a performance requirement placed on the fuel for the system (i.e., transportation or storage cask) to meet a regulation that does not specifically require performance capabilities of the SNF. Examples include:

10 CFR 71.55(e) states in part: "A package used for the shipment of fissile material must be so designed and constructed and its contents so limited that under the tests specified in 10 CFR 71.73 ('Hypothetical accident conditions'), the package would be subcritical."

Note: This regulation does not place a specific requirement on the SNF. However, if the package requires the SNF to maintain its geometric configuration to ensure subcriticality, then a function is imposed on the SNF.

10 CFR 72.122(h)(5) states: "The high level radioactive waste and reactor related GTCC waste must be packaged in a manner that allows handling and retrievability without the release of radioactive materials to the environment or radiation exposures in excess of Part 20 limits . . . "

10 CFR 72.124(a) states: "Design for criticality safety. Spent fuel handling, packaging, transfer, and storage systems must be designed to be maintained subcritical and to ensure that, before a nuclear criticality accident is possible, at least two unlikely, independent, and concurrent or sequential changes have occurred in the conditions essential to nuclear criticality safety. The design of handling, packaging, transfer, and storage systems must include margins of safety for the nuclear criticality parameters that are commensurate with the uncertainties in the data and methods used in calculations and demonstrate safety for the handling, packaging, transfer, and storage conditions and in the nature of the immediate environment, under accident conditions."

Note: If the SNF must have certain characteristics or behave in a specified manner to maintain the required margins, a function is placed on the SNF.

10 CFR 72.128 states in part: "Spent fuel storage . . . must be designed to ensure adequate safety under normal and accident conditions. These systems must be designed with (2) suitable shielding for radioactive protection under normal and accident conditions, (3) confinement structures and systems . . . "

Note: If proper functioning of the shielding or containment requires that the SNF maintain its configuration, then a function is placed on the SNF.

Applicability

This guidance applies to reviews of dry cask storage systems and transportation casks conducted in accordance with NUREG-1536, "Standard Review Plan for Dry Cask Storage Systems" (January 1997); NUREG-1567, "Standard Review Plan for Spent Fuel Dry Storage Facilities" (March 2000); or NUREG-1617, "Standard Review Plan for Transportation Packages for Spent Nuclear Fuel" (March 2000). This revision of ISG-1 supersedes any definitions of damaged, grossly damaged, or intact fuel in the above Standard Review Plans.

This revision supersedes ISG-1, Revision 1, "Damaged Fuel," in its entirety, and is applicable to both as-built and reconstituted fuel assemblies.

Definitions

1. Spent Nuclear Fuel (SNF) - See 10 CFR Part 72.3 for definition. This term has been used in the nuclear industry, at different times, to mean the fuel pellets, the rod, or entire fuel assembly. Unless specifically modified, the term will refer to both the rods and fuel assembly.

2. Damaged SNF - Any fuel rod or fuel assembly that cannot fulfill its fuel-specific or system-related functions.

3. Undamaged SNF - SNF that can meet all fuel-specific and system-related functions. As shown in Figure 1, undamaged fuel may be breached. Fuel assembly classified as undamaged SNF may have "assembly defects."

4. Breached spent fuel rod - Spent fuel rod with cladding defects that permit the release of gas from the interior of the fuel rod. A breached spent fuel rod may also have cladding defects sufficient to permit the release of fuel particulate. A breach may be limited to a pinhole leak or hairline crack, or may be a gross breach.

5. Pinhole leaks or hairline cracks - Minor cladding defects that will not permit significant release of particulate matter from the spent fuel rod, and therefore present a minimal as-low-as-is-reasonably-achievable concern, during fuel handling and retrieval operations. (See discussion of gross defects for size concerns.)

6. Grossly breached spent fuel rod - A subset of breached rods. A breach in spent fuel cladding that is larger than a pinhole leak or a hairline crack. An acceptable examination for a gross breach is a visual examination that has the capability to determine the fuel pellet surface may be seen through the breached portion of the cladding. Alternatively, review of reactor operating records may provide evidence of the presence of heavy metal isotopes indicating that a fuel rod is grossly breached. (See discussion for size concerns.)

7. Intact SNF - Any fuel that can fulfill all fuel-specific and system-related functions, and that is not breached. Note that all intact SNF is undamaged, but not all undamaged fuel is intact, since under most situations, breached spent fuel rods that are not grossly breached will be considered undamaged.

8. Can for Damaged Fuel - A metal enclosure that is sized to confine one damaged spent fuel assembly. A fuel can for damaged spent fuel with damaged spent-fuel assembly contents must satisfy fuel-specific and system-related functions for undamaged SNF required by the applicable regulations.

9. Assembly Defect - Any change in the physical as-built condition of the assembly with the exception of normal in-reactor changes such as elongation from irradiation growth or assembly bow. Examples of assembly defects: (a) missing rods; (b) broken or missing grids or grid straps (spacers); and (c) missing or broken grid springs, etc. An assembly with a defect is damaged only if it can't meet its fuel-specific and system-related functions required by the applicable regulations.

Note: See Appendix for default definition of damaged SNF.

Background

Damaged Fuel

Previous definitions of damaged fuel have identified specific characteristics of the fuel that classify it as damaged, irrespective of whether the fuel is being stored or transported and independent of the design of the storage or transportation system. In this guidance, damaged fuel is defined in terms of the characteristics needed to perform the fuel-specific and system-related functions. Thus, the characteristics of damaged spent fuel may depend on (1) whether the fuel is being stored or transported, and (2) the design of the storage or transportation system.

The materials properties, and possibly the physical condition, of a fuel rod or assembly can be altered during irradiation, storage, or transportation. If this alteration is large enough to prevent the fuel or assembly from performing its fuel-specific or system-related functions during storage, transportation, or both then the fuel assembly is considered damaged.

To determine whether a fuel assembly is undamaged, the following should be delineated:

1) Whether the definition is applicable to storage, transportation or both;

2) The functions the applicant has imposed on the fuel rods and assembly by either fuel-specific or system-related functions to meet a regulatory requirement for the designated phase (storage, transportation, or both);

3) The mechanisms of change (alteration mechanisms) or the characteristics of the fuel that could potentially cause the fuel to fail to meet its fuel-specific or system-related functions;

4) An acceptable analysis showing that the fuel with the designated characteristics will meet the fuel-specific and system-related functions when the mechanisms considered in item #3, above, are evaluated; and

5) The physical characteristics of the fuel, based on item #4, above, that could cause the fuel or assembly to be classified as "damaged."

The "Discussion" section illustrates this methodology in the example.

Damaged SNF, as defined in this guidance, will only be approved for the activity (storage, transportation, or both) for which the application is being submitted. Note that the "default" definition of damaged SNF, derived from ANSI N14.33-2005, is provided in the appendix of this

guidance for those that do not want to perform the assessment outlined in item numbers 1 through 5 above [2]. The default definition, however, may not take full advantage of the flexibility of the performance-based definition of damaged fuel provided in this guidance. This default definition may be more restrictive than necessary, depending on the design of the storage or transportation cask. For example, the default definition of damaged SNF indicates that SNF must be classified as damaged if an individual fuel rod is missing from an assembly. However, if an analysis shows that all fuel-specific and system-related functions will be met (e.g., subcriticality will be maintained, that the SNF assembly will be retrievable and that the structural properties of the assembly are not compromised by the missing rod) the assembly may be classified as undamaged, per this ISG.

Discussion

The performance-based definition [3,4] of damaged SNF provided in this ISG minimizes the quantity of damaged fuel requiring alternative handling paths, while still addressing applicable system-related regulations concerning criticality control, thermal limitations, structural integrity, confinement, and shielding.

A. Grossly Breached SNF Cladding

The regulations in 10 CFR 72.122(h) state "The spent fuel cladding must be protected during storage against degradation that leads to gross ruptures or the fuel must be otherwise confined such that degradation of the fuel during storage will not pose operational safety problems with respect to its removal from storage." However, there is no such requirement in 10 CFR 71. Hence, grossly breached fuel is always considered damaged for storage, but may, or may not, be considered damaged for the purposes of transportation depending on whether other regulations, such as criticality, can be met.

In dry cask storage and transportation systems, a gross cladding breach should be considered as any cladding breach that could lead to the release of fuel particulate greater than the average size fuel fragment. A pellet is ~1.1 centimeters in diameter in 15 x 15 Pressurized-Water Reactor (PWR) assemblies. Pellets from a Boiling-Water Reactor (BWR) are somewhat larger, and those from 17 x 17 PWR assemblies are somewhat smaller. The pellet's length is slightly longer than its diameter. During the first cycle of irradiation in-reactor, the pellet fragments into 25-35 smaller interlocked pieces, plus a small amount of finer powder, due to, pellet-to-pellet abrasion. When the rod breaches, about 0.1 gram of this fine powder may be carried out of the fuel rod at the breach site [5]. Modeling the fragments as either spherical- or pie-shaped pieces indicates that a cladding-crack width of at least 2-3 millimeters would be required to release a fragment. Hence, gross breaches should be considered to be any cladding breach greater than 1 millimeter.

A review of reactor operating records, ultrasonic testing, and sipping (if done in a timely fashion) can be used to classify rods as unbreached or, breached. Evidence of only gaseous or volatile decay products (no heavy metals) in the reactor coolant system is accepted as evidence that a cladding breach is no larger than a pinhole leak or hairline crack. Records that show the presence of heavy metal isotopes that are characteristic of fuel release in the reactor coolant system indicate gross breaches in the cladding. Likewise, visual examination may also be used to determine if a cladding breach is gross, if the breached rod can be positively identified.

Because cladding openings larger than 1 millimeter should expose the fuel pellet to visual sighting, visual examination of the breached rod can be used to determine if a breach is gross. However, visual examination is not an acceptable method of confirming intact (undamaged) fuel for assemblies that have indicated leakage.

It should be noted; however, that undamaged spent-fuel rods with pinhole leaks and/or hairline cracks will expose the fuel pellets to the canister or cask atmosphere. If that atmosphere is oxidizing, then the fuel pellet may oxidize and expand, placing stress on the cladding. The expansion may eventually cause a large split in the cladding, resulting in spent fuel that must be classified as damaged (for storage and possibly also for transportation) due to gross breaches in the cladding. Since fuel oxidation and cladding splitting follow Arrhenius time-at-temperature behavior, fuel rods with pinholes or hairline cracks that are exposed to an oxidizing atmosphere may experience this type of additional cladding damage. ISG-22 “Potential Rod Splitting Due to Exposure to an Oxidizing Atmosphere During Short-Term Cask Loading Operations in LWR or other Uranium Oxide Based Fuels” [6] provides information regarding prevention of this phenomenon. Before handling undamaged rods with pinhole leaks and/or hairline cracks in an oxidizing atmosphere, the potential fuel and cladding degradation at the temperature of interest for the duration of the process should be assessed.

B. Fuel Assembly with Defects

Damage under this guidance refers to alterations of the fuel assembly that prevent it from fulfilling its fuel-specific or system-related functions. Defects such as dents in rods, bent or missing structural members, small cracks in structural members, missing rods, etc., need not be considered damaged if the applicant can show that the fuel assembly with these defects still fulfils its fuel-specific and system-related functions. This may be done using calculations based on approved codes, situation-specific data, or reasoned engineering arguments.

C. Canning Damaged Fuel

Spent fuel that has been classified as damaged for storage must be placed in a can designed for damaged fuel, or in an acceptable alternative. The purpose of a can designed for damaged fuel is to (1) confine gross fuel particles, debris, or damaged assemblies to a known volume within the cask; (2) to demonstrate that compliance with the criticality, shielding, thermal, and structural requirements are met; and (3) permit normal handling and retrieval from the cask. The can designed for damaged fuel may need to contain neutron-absorbing materials, if results of the criticality safety analysis depend on the neutron absorber to meet the requirements of 10 CFR 72.124(a).

D. Relationship of Spent Fuel Populations

The applicant will designate the population of spent fuel for which the cask system was designed (e.g., type of fuel, minimum cooling time, burnup limitations, arrays, manufacturers, cladding types, etc.) This population may contain breached rods. Some of these breached rods may be grossly breached. It may also contain assemblies with defects, such as missing rods, missing grid spacers, or damaged spacers. The populations of breached rods, grossly breached rods, and assemblies with defects are determined by in-reactor behavior and ex-reactor handling.

Each of these populations must be classified as damaged or undamaged after the storage or transportation system has been designated. For example, an applicant might propose the use of air as a cover gas in its design of a storage cask. The applicant might also propose this cask for use in storing spent fuel with cladding breaches that are hairline cracks or pinhole leaks. However, if the spent fuel in the cask will operate at a sufficiently high temperature for a long enough time, then oxidation of fuel pellets in breached rods could occur resulting in gross breaches. If this is the case, the breached spent fuel should be considered damaged because grossly breached rods do not meet the requirements of 10 CFR 72.122(h)(1). Also, in this case because the geometric form of the package contents could be substantially altered, the spent fuel would also be classified as damaged for transportation because the requirements of 10 CFR 71.55(d)(2) might not be met. If an inert atmosphere was used instead of air, only grossly breached rods would be considered damaged for storage. This concept is illustrated in Figure 1, "Relationship of Spent Fuel Populations."

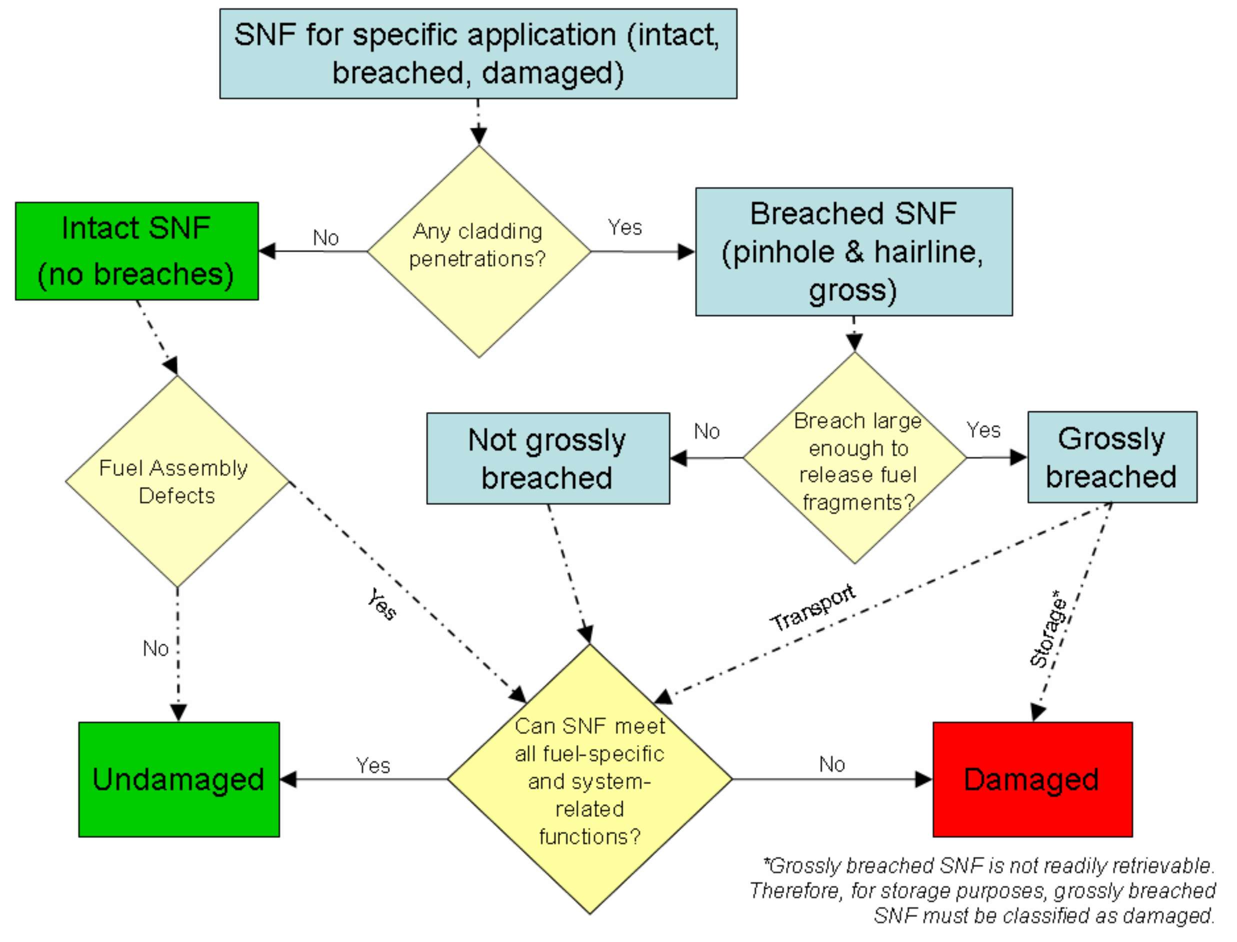

Figure 1 - Relationship of Spent Fuel Populations

E. Example of Methodology

The following example is given to illustrate the general methodology adopted in this ISG. This is only an example of the methodology and should not be construed as approved characterization of damaged fuel.

Example:

Situation - The vendor of a dual-purpose cask wants to store and transport low-burnup PWR fuel in an inert atmosphere and within the temperature limits recommended in ISG-11, Revision 3, “Cladding Considerations for the Transportation and Storage of Spent Fuel” [7]. The vendor wants to store assemblies having rods with breaches containing only pinholes or hairline cracks, and assemblies having one or more outer grid straps with defects at three or more grid locations without canning them. The vendor is only applying for a storage license at this time but wants to be reasonably certain that the fuel will also be transportable.

Activity - Storage and transportation

Fuel-specific or system-related functions imposed on rods and assemblies - 10 CFR 72.122(h)(1), regarding gross ruptures, and 10 CFR 72.122(l), concerning retrievability, must be met for storage. 10 CFR 71.55(d), requiring the system to remain subcritical and unchanged during normal transport, must be met. The vendor believes that all the remaining system requirements, except for the subcriticality requirement, can be met, without imposing any limitations on the fuel, if the fuel is within the bounds stated in the situation.

Mechanisms - There are no mechanisms for the pinhole leaks and hairline cracks to evolve into gross breaches since the atmosphere is inert and the temperature is controlled. To be retrievable, the assemblies with missing grid straps must be able to withstand design basis events in a storage cask. Since the applicant also wants these assemblies to be considered undamaged for transportation, the behavior of the assemblies under both normal and hypothetical accident transportation conditions in 10 CFR 71 must be evaluated. For example, for normal transportation conditions, the applicant must show that the assemblies with the most missing grid straps in the worst locations can withstand both normal vibration and a one-foot drop and remain in their original physical configuration. Additionally, for hypothetical accident conditions, the analysis must indicate, among other things, that the system will meet shielding and subcriticality requirements when placed under the mechanical and thermal loads specified in 10 CFR 71.

Analysis - The applicant conducts an analysis to satisfactorily demonstrate that the assembly with three missing grid straps in the worst configuration remains intact for 1) normal transportation conditions; 2) cask tip-over; and 3) regulatory accident conditions. Further acceptable analysis indicates that all the system-related regulations are met, if the fuel with the characteristic limitations (as noted in Characteristics section below), stays structurally intact.

Characteristics - Assemblies containing breached rods with up to three grid straps missing will be considered undamaged for the purposes of storage. Analysis shows that these assemblies could probably also be considered undamaged for transportation, but fuel with these characteristics will be evaluated and approved as part of a later application for the transportation cask certification.

Records

Records documenting the classification of spent fuel shall comply with the provisions of 10 CFR 72.174, “Quality Assurance Records”; 10 CFR 72.72, “Material Balance, Inventory, and Records Requirements for Stored Material”; 10 CFR 71.91, “Records”; and 10 CFR 71.135, “Quality Assurance Records.” Inspection records will be maintained for the lifetime of the container.

Quality Assurance

Activities related to inspection, evaluation, and documentation of damaged spent fuel for dry storage shall be performed in accordance with a quality assurance program, as required in 10 CFR Part 72, Subpart G, “Quality Assurance.” Activities related to inspection, evaluation, and documentation of damaged spent fuel for transport shall be performed in accordance with a quality assurance program, as required in 10 CFR Part 71, Subpart H, “Quality Assurance.”

Recommendations

The staff recommends that: (1) the definitions in NUREG-1536, NUREG-1567, NUREG-1617, ISG-8, Revision 2 and ISG-11, Revision 3 be revised to incorporate the definitions listed above; and (2) the appropriate chapters of each NUREG be revised to include the discussion section of this ISG.

In addition, the suggestion in NUREG-1617 (canning damaged fuel is necessary for the purposes of transportation) should be modified to be consistent with this ISG, nothing that canning damaged fuel may, in some cases, be necessary to meet the requirements of 10 CFR Part 71.

The words “intact fuel,” in the Applicability section of Revision 2 of ISG-8, “Burnup Credit in the Criticality Safety Analyses of PWR Spent Fuel in Transport and Storage Casks,” should be replaced with “undamaged fuel.” “Intact commercial spent fuel" in the last paragraph of the “Issue” section of Revision 3 of ISG-11, “Cladding Considerations for the Transportation and Storage of Spent Fuel,” should read “undamaged commercial spent fuel.” The first sentence of the Issue Section of ISG-22 should be modified to be consistent with the current definitions. “Rev 1,” should be changed to “Rev 2,” with the new title to ISG-1 Rev 2 and “intact fuel” should be changed to “unbreached fuel.”

Approved :

/RA/	May 11, 2007
E. William Brach, Director Division of Spent Fuel Storage and Transportation	Date

Attachment: Appendix

APPENDIX

Default Definition of Damaged Fuel[2]

"Default" definition of damaged Spent Nuclear Fuel (SNF) - SNF assemblies must be classified as damaged, for both dry storage and/or transportation purposes, if any one of the following conditions exist:

On removal of SNF selected for dry storage or transport from the spent fuel pool, any of the following apply:

1. There is visible deformation of the rods in the SNF assembly. Note: This is not referring to the uniform bowing that occurs in the reactor. This refers to bowing that significantly opens up the lattice spacing.

2. Individual fuel rods are missing from the assembly. Note: The assembly may be reclassified as intact if a dummy rod that displaces a volume equal to, or greater than, the original fuel rod, is placed in the empty rod location.

3. The SNF assembly has missing, displaced, or damaged structural components such that either:

    3.1 Radiological and/or criticality safety is adversely affected (e.g., significantly changed rod pitch).

    3.2 The assembly cannot be handled by normal means (i.e., crane and grapple).

4. Reactor operating records (or other records) indicate that the SNF assembly contains fuel rods with gross breaches.

5. The SNF assembly is no longer in the form of an intact fuel bundle (e.g., consists of, or contains, debris such, as loose fuel pellets or rod segments).

[2]Derived from ANSI Standard N14.33-2005, "Storage and Transport of Damaged Spent Nuclear Fuel," September 2005.

REFERENCES

1. NRC Spent Fuel Project Office Interim Staff Guidance - 2, "Fuel Retrievability."

2. ANSI Standard N14.33 2005, "Storage and Transport of Damaged Spent Nuclear Fuel," September 2005.

3. RE Einziger, CL Brown, GP Hornseth, SR Helton, NL Osgood, and CG Interrante, "Damage in Spent Nuclear Fuel Defined by Properties and Requirements," Presented at IAEA Technical Workshop on Damaged Fuel, Dec 2005, Vienna, Austria, Proceedings of IAEA International Conference on Management of Spent Fuel from Nuclear Power Reactors, Vienna, Austria, June 2006.

4. MW Hodges, "Status of Technical Issues in Storage and Transportation," Proceedings of Spent Fuel Management Seminar XXIII, Washington, D.C., Institute for Nuclear Materials Management, January 2006.

5. RA Lorenz, et al., "Fission Product Release from BWR Fuel Under LOCA Conditions," Oak Ridge National Laboratory, Oak Ridge, TN, NUREG/CR-1773, July 1981.

6. NRC Spent Fuel Project Office Interim Staff Guidance - 22, "Potential Rod Splitting Due to Exposure to an Oxidizing Atmosphere During Short Term Cask Loading Operations of LWR or Other Uranium Oxide Based Fuels," May 2006.

7. NRC Spent Fuel Project Office Interim Staff Guidance - 11, Rev 3, "Cladding Considerations for Transportation and Storage of Spent Fuel," November 2003.

CERTIFICATE OF SERVICE

I hereby certify that on July 20, 2020, I electronically filed the foregoing with the Clerk of the Court for the United States Court of Appeals for the Ninth Circuit by using the appellate CM/ECF system.

Participants in the case who are registered CM/ECF users will be served by the appellate CM/ECF system.

Dated: July 20, 2020

Respectfully submitted,
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