No. 20-70899

UNITED STATES COURT OF APPEALS
FOR THE NINTH CIRCUIT

PUBLIC WATCHDOGS,
Petitioner,

v.

UNITED STATES NUCLEAR REGULATORY COMMISSION and
UNITED STATES OF AMERICA,
Respondents,

SOUTHERN CALIFORNIA EDISON COMPANY,
Intervenor.

On Petition for Review of an Order of
the U.S. Nuclear Regulatory Commission

FEDERAL RESPONDENTS' SUPPLEMENTAL EXCERPTS OF RECORD

JEFFREY BOSSERT CLARK
Assistant Attorney General
ERIC GRANT
Deputy Assistant Attorney General
JUSTIN D. HEMINGER
EVELYN S. YING
Attorneys
Environment and Natural Resources Division
U.S. Department of Justice
evelyn.ying@usdoj.gov
(202) 514-5786

MARIAN L. ZOBLER
General Counsel
ANDREW P. AVERBACH
Solicitor
JAMES E. ADLER
Senior Attorney
Office of the General Counsel
U.S. Nuclear Regulatory Commission
andrew.averbach@nrc.gov
(301) 415-1956

FEDERAL RESPONDENTS' SUPPLEMENTAL EXCERPTS OF RECORD: TABLE OF CONTENTS

Documents from NRC's Certified Index of the Record (Dkt. 27)

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

CONTACT: Shawn W. Harwell, NMSS/REFS
(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.

<u>DISCUSSION</u>:

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

---

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.

---

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.

<u>CONCLUSION</u>:

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.

---

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

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

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. Specifically, 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 <u>(Closed) NRC Event Notification #53858, Inadequate Analysis for VCT Operations</u>

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 <u>Finding related to the Licensee's Event Notification</u>

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
<u>Discussed</u>		
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

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

SOUTHERN CALIFORNIA
EDISON
An EDISON INTERNATIONAL® Company

Doug Bauder
Chief Nuclear Officer and
Vice President, Decommissioning

EA-18-155

April 23, 2019

Director, Office of Enforcement
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

SUBJECT: **Docket Nos. 50-206, 50-361, 50-362 and 72-41**
Reply to a Notice of Violation, EA-18-155, and
Statement of Method of Payment
San Onofre Nuclear Generating Station (SONGS),
Units 1, 2, 3, and ISFSI

REFERENCES:

1. Letter from Mr. Troy Pruett (NRC) to Doug Bauder (SCE) dated November 28, 2018, Subject: 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, (ADAMS Accession No. ML18332A357)

2. Southern California Edison Company; San Onofre Nuclear Generating Station, Pre-Decisional Enforcement Conference Slides, dated January 24, 2019 (ADAMS Accession No. ML19023A033)

3. Letter from Scott A. Morris (NRC) to Doug Bauder (SCE) dated March 25, 2019, Subject: Notice of Violation and Proposed Imposition of Civil Penalty - $116,000 and NRC Special Inspection Report 050-00206/2018-005, 050-00361/2018-005, 050-00362/2018-005, 072-00041/2018-001 (ADAMS Accession No. ML19080A208)

OEØ1
IE01
IE08
NMSS26
RGN-IV
NMSS
NRC/OE

Dear Sir or Madam

Reference 1 transmitted the results of NRC Special Inspection Report Numbers 050-00206/2018-005, 050-00361/2018-005, 050-00362/2018-005, and 072-00041/2018-001 to Southern California Edison (SCE). The inspection was conducted on-site from September 10, 2018 to September 14, 2018 for the San Onofre Nuclear Generating Station (SONGS). The inspection was in response to the misalignment of a loaded spent fuel storage canister as it was being downloaded into the storage vault at SONGS. Reference 1 discussed two apparent violations that were under consideration for escalated enforcement and provided SCE options for responding. SCE selected a Pre-Decisional Enforcement Conference which was held on January 24, 2019 (Reference 2).

Reference 3 issued a Notice of Violation resulting from the two issues discussed in Reference 1 and a Proposed Imposition of Civil Penalty to SCE. Reference 3 required SCE to reply to

the violations in writing within 30 days. Reference 3 also required SCE to provide a statement indicating when and by what method payment was made or to protest the imposition of the civil penalty in whole or in part.

The enclosure to this letter provides SCE's acceptance of and reply to the Notice of Violation provided in Reference 3.

In addition, SCE has chosen to pay the proposed Civil Penalty of $116,000 and will not contest the Civil Penalty in whole or in part. SCE made the required payment to the NRC via wire payment on March 29, 2019.

There are no new regulatory commitments in this letter or the Enclosure.

If you have any questions or require additional information, please contact me or Mr. Albert Bates, at (949) 368-6945.

Executed on 4/23/19

Sincerely,

Enclosure: Reply to Notice of Violation EA-18-155

cc: Document Control Desk
S. Morris, Regional Administrator, NRC Region IV
M. Vaaler, NRC Project Manager, SONGS Units 1, 2 and 3

ENCLOSURE

Reply to Notice of Violation

EA-18-155

Reply to Notice of Violation

EA-18-155

BACKGROUND AND EVENT SUMMARY

On January 22nd, 2018, the San Onofre Nuclear Generating Station (SONGS) began a fuel transfer campaign from the Units 2 and 3 spent fuel pools to a Holtec HI-STORM UMAX Independent Spent Fuel Storage Installation (ISFSI). The campaign will ultimately result in the transfer of 73 canisters to the ISFSI.

On Friday, August 3, 2018, at approximately 12:45 PDT, workers at SONGS were lowering Multi-Purpose Canister (MPC) number 29 into a Cavity Enclosure Container (CEC) within the Independent Spent Fuel Storage Installation (ISFSI). Workers used a Vertical Cask Transporter (VCT) to perform the download operation. As the MPC was lowered, it came to rest on top of the shield ring and against the inner wall of the transfer cask. The VCT slings went slack indicating the MPC was hung up.

The VCT operator could not see the MPC as it was being lowered within the transfer cask. The spotter assigned to observe the MPC did not recognize the slack sling condition. The Cask Loading Supervisor (CLS), Rigger-in-Charge (RIC) and the Southern California Edison (SCE) ISFSI Project Oversight Specialist were located 150 feet away in a low radiation dose area and did not have a visible way to monitor the lowering of the MPC into the CEC.

With the MPC supported by the shield ring, the crane and rigging no longer supported it. Dose rate measurements taken near the VCT indicated that the MPC had not been lowered to its fully downloaded condition. Actions were taken immediately to raise the VCT, regaining support of the MPC by the VCT. The MPC was then safely lowered past the shield ring and into storage at 14:14 PDT.

At that time, MPC number 30 was being prepared for transfer to dry storage in the SONGS Unit 3 Fuel Handling Building. The MPC was seismically restrained in the Unit 3 Fuel Handling Building and then closure welding was completed. Since that time, SONGS has suspended all fuel movement pending completion of cause evaluations and required corrective actions. SCE will not re-start fuel transfer operations until the NRC has reviewed SCE's corrective actions and SCE management is satisfied full readiness has been achieved to ensure safe and effective fuel transfer operations.

This event was informally communicated to NRC Region IV on Monday, August 6, 2018. A late report was made to the NRC Headquarters Operations Center on Friday, September 14, 2018 in accordance with 10 CFR 72.75(d)(1).

Following the event, Holtec completed a root cause evaluation in accordance with its corrective action program to determine causes and appropriate corrective actions to prevent recurrence. SCE reviewed and accepted Holtec's root cause evaluation. In addition, SCE completed an Apparent Cause Evaluation to examine how SCE's oversight failed to prevent the event and a Common Cause Evaluation to examine issues related to fuel transfer in the

areas of administration, and problem identification and resolution. SCE's Apparent Cause Evaluation and Common Cause Evaluation also resulted in corrective actions to prevent recurrence of these problems. SCE also completed a Root Cause Evaluation to determine the causes of the late report to the NRC Headquarters Operations Center.

On March 25, 2019, the NRC issued a Notice of Violation (NOV) and Proposed Imposition of Civil Penalty numbered EA-18-155 and requested a reply within 30 days. The reply to the Notice of Violation appears below.

DESCRIPTION OF VIOLATION 18-155-A

10 CFR 72.172(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. 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, in part, 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 Specifications, Appendix A, Section 5.2.c.3 requires, in part, 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 the redundant drop protection features were available to prevent uncontrolled lowering of the load during multi-purpose canister transfer operations. 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 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 final safety analysis report.

SCE REPLY TO VIOLATION 18-155-A

1. Reason for the Violation

   The Root Cause Evaluation identified a Root Cause that Holtec Management failed to recognize the complexity and risks associated with fuel transfer operation while using a relatively new system design (UMAX) when performing a long duration campaign and thus did not implement necessary program improvements. Contributing causes included:

   - Inadequate procedure content
   - Inadequate design review process
   - Poor communication protocols
   - A Continuous Learning Environment not established for use of Operating Experience
   - Inadequate training program

The Apparent Cause Evaluation identified an Apparent Cause of SCE failing to establish a rigorous oversight process. Contributing causes included:

- Project Management Observations not routinely performed
- A low threshold for Corrective Action Program entries not enforced

2. Corrective Actions Taken and Results Achieved

   Corrective actions related to procedure content and control, training and qualifications, and the use of the Corrective Action Program (CAP) were described in detail in SCE's reply to a Notice of Violation dated December 26, 2018 (ADAMS Accession No. ML18362A148). In addition, corrective actions related to each of the causes identified in item 1, above, are described in SCE's presentation for the January 24, 2019 Pre-Decisional Enforcement Conference (ADAMS Accession No. ML19023A033), slides 39 – 47.

   Finally, as an additional corrective action from the Apparent Cause Evaluation, SCE has taken action to require enhanced load monitoring equipment, including load monitoring shackles with remote indication and alarms, cameras and monitors installed to observe downloading remotely, and a tag-line indicator installed on the MPC for physical verification of downloading.

3. Corrective Actions That Will Be Taken

   None.

4. Date When Full Compliance Will Be Achieved

   Full compliance was achieved on August 3, 2018, when the VCT regained support of the MPC.

DESCRIPTION OF VIOLATION 18-155-B

10 CFR 72.75(d)(1) requires, in part, that each licensee shall notify the NRC within 24 hours after the discovery of an event involving spent fuel in which important-to-safety equipment is disabled or fails to function as designed when: (i) the equipment is required by Certificate 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 within the required time period after the discovery of an event involving spent fuel in which important-to-safety equipment was disabled or failed to function as designed when: (i) the equipment was required by Certificate 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.

Specifically, the licensee failed to notify the NRC within the required time period after an event that occurred on August 3, 2018, in which the licensee inadvertently disabled the redundant important-to-safety downloading slings while lowering spent fuel canister 29 into the storage vault, which resulted in the canister resting on a shield ring unsupported by the redundant downloading slings at approximately 18 feet above the fully seated position for approximately 45 minutes. These slings are required by Certificate of Compliance 072-01040, Amendment

2, Condition 4, and Technical Specification 5.2.c.3 to be available and operable during canister transfer operations, and no redundant equipment was available and operable to perform the required safety function.

SCE REPLY TO VIOLATION 18-155-B

1. Reason for the Violation

   The Root Cause Evaluation on Reportability identified a root cause that management failed to recognize the transition to fuel transfer operations as requiring the integration, familiarization, and application of 10CFR72.75 reporting requirements into plant processes. Contributing causes included:

   - Lack of procedural guidance related to Part 72 reporting
   - Lack of a conservative bias for reporting

2. Corrective Actions Taken and Results Achieved

   Corrective actions related to each of the causes identified in item 1, above, are described in SCE's presentation for the January 24, 2019 Pre-Decisional Enforcement Conference (ADAMS Accession No. ML19023A033), slides 60 – 67.

3. Corrective Actions That Will Be Taken

   None.

4. Date When Full Compliance Will Be Achieved

   Full compliance was achieved on September 14, 2018, when the required report was submitted.

U.S. NUCLEAR REGULATORY COMMISSION MANAGEMENT DIRECTIVE (MD)

MD 8.11	REVIEW PROCESS FOR 10 CFR 2.206 PETITIONS	DT-19-01
Volume 8:	Licensee Oversight Programs	
Approved By:	Margaret M. Doane, Executive Director for Operations	
Date Approved:	March 1, 2019	
Cert. Date:	N/A, for the latest version of any NRC directive or handbook, see the online MD Catalog	
Issuing Office:	Office of Nuclear Reactor Regulation Division of Operating Reactor Licensing	
Contact Name:	Perry Buckberg	

EXECUTIVE SUMMARY

Management Directive (MD) 8.11, “Review Process for 10 CFR 2.206 Petitions,” is being revised to—

- Clarify the initial screening and acceptance criteria for evaluating petitions,
- Clarify guidance regarding coordination and referral of allegations,
- Clarify and update roles and organizational responsibilities,
- Clarify and add guidance regarding referrals from adjudicatory boards and the Commission,
- Clarify guidance on public meeting and teleconference interactions,
- Clarify guidance for a streamlined director’s decision in certain cases,
- Correct the addressee of the periodic 2.206 status report from the Commission to the Director of the Office of Nuclear Reactor Regulation,
- Revise the process to accelerate the PRB initial assessment prior to meeting with the petitioner,
- Add a timeliness goal for issuing the acknowledgment or closure letter,
- Add criteria for holding a petition in abeyance,
- Clarify that the PRB chairperson is the final decision maker for the PRB,
- Add guidance on requests to impose requirements outside of NRC jurisdiction,
- Add the Office of International Programs to the offices responsible for petitions, and
- Relocate detailed procedural staff guidance to “Desktop Guide: Review Process for 10 CFR 2.206 Petitions,” to clarify and facilitate future updates, as needed.

For updates or revisions to policies contained in this MD that were issued after the MD was signed, please see the Yellow Announcement to Management Directive index (YA-to-MD index).

TABLE OF CONTENTS

I. POLICY

It is the policy of the U.S. Nuclear Regulatory Commission (NRC) to provide any person with the means to request that the NRC institute a proceeding pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) 2.202, “Orders,” to modify, suspend, or revoke a license, or for other action as may be proper (hereinafter referred to in this directive as to take enforcement-related action). This policy is codified in 10 CFR 2.206, “Requests for Action Under This Subpart.” The NRC may grant a request for action, in whole or in part, take other action that satisfies the concerns raised by the requester, or deny the request. Requests

that raise health and safety and other concerns without requesting enforcement-related action will be reviewed by means other than the 10 CFR 2.206 process.

II. OBJECTIVES

– Ensure public health and safety through the prompt and thorough evaluation of any potential problem addressed by a petition filed under 10 CFR 2.206.

– Provide for appropriate participation by a petitioner in the NRC's decisionmaking activities related to a 10 CFR 2.206 petition.

– Ensure effective communication with the petitioner and other stakeholders on the status of a petition, including providing relevant documents and notification of interactions between NRC staff and a licensee or certificate holder relevant to the petition.

III. ORGANIZATIONAL RESPONSIBILITIES AND DELEGATIONS OF AUTHORITY

A. Executive Director for Operations (EDO)

Receives and assigns action for all petitions filed under 10 CFR 2.206.

B. Office of the General Counsel (OGC)

1. Provide legal advice to the Commission, EDO, office directors, and staff on matters related to the 10 CFR 2.206 process.
2. Provide legal counsel on matters related to the 10 CFR 2.206 petition process, upon specific request from the staff in a special case or where a petition raises legal issues. Reviews written correspondence between the staff and the petitioner(s) such as letters and staff decisions (e.g., proposed and final director's decisions).

C. Director, Office of Enforcement (OE)

1. Provides enforcement and allegation program advice to the Commission, EDO, office directors, and staff on matters related to the 10 CFR 2.206 process
2. Provides enforcement and allegation program advice on a 10 CFR 2.206 petition submittal and, upon specific request from the staff, reviews written correspondence between the staff and the petitioner(s) such as letters and staff decisions (e.g., proposed and final director’s decisions).

D. Director, Office of Investigations (OI) and Inspector General (IG)

1. The Office of Investigations (OI) provides advice on a 10 CFR 2.206 petition submittal upon specific request from the staff in a special case or where a petition raises any allegation of wrongdoing by a licensee or certificate holder, applicant for a licensee or certificate, their contractor, or their vendor.

2. The Office of the Inspector General (OIG) addresses suspected wrongdoing by NRC employees and contractors such as mismanagement of agency programs that could adversely impact matters related to public health and safety.

3. Any mention outside the NRC of an ongoing OI or OIG investigation requires the approval of the Director of OI or the IG, respectively.

E. Director, Office of Nuclear Reactor Regulation (NRR)

1. Responsible for the development and implementation of agencywide policy and procedures regarding the processing of 10 CFR 2.206 petitions.

2. For assigned petitions, see additional roles and responsibilities in Section III.F of this directive.

F. Directors, Office of Nuclear Reactor Regulation (NRR), Office of New Reactors (NRO), Office of Nuclear Material Safety and Safeguards (NMSS), and Office of International Programs (OIP)

1. Responsible for an assigned petition. Because 10 CFR 2.206 petitions request enforcement-related action against entities licensed or otherwise regulated by the NRC, petitions are assigned to the Office of Nuclear Reactor Regulation (NRR), Office of New Reactors (NRO), Office of Nuclear Material Safety and Safeguards (NMSS), and Office of International Programs (OIP).

2. Designate an office 2.206 petition coordinator.

3. Approve or deny staff decisions to take immediate action on issues raised in a 2.206 petition.

4. Concur on closure letters and letters transmitting proposed director's decisions for comment.

5. Sign acknowledgment letters and associated *Federal Register* notices of receipt.

6. Sign director's decisions.

7. For each petition, establish a process to appoint or re-delegate to the appropriate staff the following:

   (a) Provide up-to-date information on all assigned petitions.

   (b) Designate the organization and staff responsible for an assigned petition, including,

      (i) A petition review board (PRB) chairperson;

      (ii) Petition manager; and

      (iii) The signature authority, typically a senior executive service (SES) manager, for letters transmitting proposed director's decisions for comments.

(c) Request OGC involvement, where appropriate, through the Assistant General Counsel for Materials Litigation and Enforcement.

(d) Request OE involvement, where appropriate.

8. Promptly notify—

(a) OI when a petition contains any allegation of wrongdoing by a licensee or certificate holder, applicant for a license or certificate, their contractor, or their vendor; and

(b) OIG when a petition contains any allegation of wrongdoing by an NRC employee or NRC contractor.

G. Regional Administrators

1. As needed, provide support and information for the preparation of an acknowledgment letter and a director's decision on a 2.206 petition.
2. Make the petition manager aware of information that is received or that is the subject of any correspondence relating to a pending petition.
3. Participate, as necessary, in meetings with the petitioner and public, in technical review of petitions and in deliberations of the PRB.

H. Deputy Office Directors, Office of Nuclear Reactor Regulation (NRR), Office of New Reactors (NRO), Office of Nuclear Material Safety and Safeguards (NMSS), and Office of International Programs (OIP)

1. Concur on PRB final recommendations.
2. Concur on PRB decisions to consolidate similar petitions or to hold a petition in abeyance.

I. Director, Division of Operating Reactor Licensing (DORL), NRR

1. Appoints the agency 2.206 petition coordinator, normally a project manager from NRR/DORL.
2. Signs the 2.206 status reports.

J. 2.206 Petition Review Board (PRB) Chairperson

Each office that is assigned a petition will appoint a PRB chairperson, generally a SES manager, who—

1. Convenes PRB meetings.
2. Is the decision maker for the PRB.

3. Ensures appropriate review of a petition in a timely manner.
4. Ensures appropriate documentation of PRB meetings.
5. Signs closure letters.

K. Agency 2.206 Petition Coordinator

1. Provides support to each office 2.206 petition coordinator to ensure consistency in implementing the 2.206 process throughout the agency.
2. Prepares a 2.206 status report, which is posted to the NRC public Web site.
3. Serves as office 2.206 petition coordinator for NRR and performs the duties listed in Section III.L of this directive.
4. Responsible for coordinating with the Office of the Secretary (SECY) in assigning director's decision numbers and informing SECY when a director's decision is signed.
5. Ensures that a periodic 2.206 program self-assessment is performed.
6. Responsible for developing and maintaining agency guidance for implementing the policy documented in MD 8.11.

L. Office 2.206 Petition Coordinator

Each office that is assigned petitions will assign an office 2.206 petition coordinator. The office 2.206 petition coordinator for each office—

1. Tracks the status of each petition within the office.
2. Coordinates the office-specific implementation of the policy documented in MD 8.11.
3. Serves on the PRB and provides advice to the PRB on implementing the 2.206 process in accordance with MD 8.11 and guidance for timely resolution.
4. Provides support to assigned 2.206 petition managers.
5. Provides the current status of petitions assigned to the office, upon request, to the agency 2.206 petition coordinator.
6. Provides guidance to staff who receive requests for enforcement-related action that are not explicitly identified as petitions under 10 CFR 2.206.
7. Convenes periodic PRB meetings with petition managers to discuss the status of open petitions and to provide guidance for timely resolution.

M. 2.206 Petition Manager

Each office that is assigned a petition assigns a 2.206 petition manager. The assigned petition manager—

1. If necessary, informs his or her office 2.206 petition coordinator of receipt of a 10 CFR 2.206 petition.
2. Performs initial screening of 10 CFR 2.206 petitions in accordance with Section II of this directive handbook.
3. Informs the office allegations coordinator and the appropriate regional allegations coordinator of a petition that involves a potential allegation.
4. Serves as the NRC point of contact for the petitioner.
5. Contacts the petitioner to determine if he or she wants the request processed as a 10 CFR 2.206 petition and determines the correct process for any petition.
6. Identifies staff members to serve on the PRB.
7. Schedules PRB meetings.
8. Prepares a written summary of the internal PRB meetings for the PRB members' review, if requested by the PRB chairperson.
9. Prepares all PRB and agency decisions and notices on 2.206 petitions in accordance with this directive handbook.
10. Provides the current status of a petition, upon request, to the office and/or agency 2.206 petition coordinator.
11. Provides any comments received on a proposed director's decision to the office 2.206 petition coordinator.
12. Prepares extension requests for review and approval in accordance with office or OEDO procedures.
13. Coordinates with the office 2.206 petition coordinator and the agency 2.206 petition coordinator when a director's decision number is needed and when the director's decision is signed.

IV. APPLICABILITY

The policy and guidance in this directive and handbook apply to all NRC employees.

V. DIRECTIVE HANDBOOK

Directive Handbook 8.11 details the procedures for staff review and disposition of a petition submitted in accordance with 10 CFR 2.206.

VI. DEFINITIONS

10 CFR 2.206 Petition

A written request filed by any person to institute a proceeding pursuant to Section 2.202 to modify, suspend, or revoke a license, or for other action as may be proper (hereinafter referred to in this directive as to take enforcement-related action).The request must meet the criteria for accepting petitions for review under 10 CFR 2.206 (see Section III.C, “Criteria for Petition Evaluation,” of this directive handbook).

Licensee

Throughout this MD, any references to a licensee shall be interpreted to include all licensees, certificate holders, and permit holders; applicants for licenses, certificates or permits; or other persons subject to the jurisdiction of the Commission.

VII. REFERENCES

Code of Federal Regulations

10 CFR 2.201, “Notice of Violation.”

10 CFR 2.202, “Orders.”

10 CFR 2.206, “Requests for Action Under This Subpart.”

10 CFR 2.390, “Public Inspections, Exemptions, Requests for Withholding.”

10 CFR 2.802, “Petition for Rulemaking.”

Nuclear Regulatory Commission Documents

Allegation Manual:
https://www.nrc.gov/about-nrc/regulatory/allegations-resp.html.

Management Directives—

3.5, “Attendance at NRC Staff-Sponsored Meetings.”

7.4, “Reporting Suspected Wrongdoing and Processing OIG Referrals.”

8.4, “Management of Facility-Specific Backfitting and Information Collection.”

8.8, “Management of Allegations.”

Guidance for Electronic Submissions to the NRC:
https://www.nrc.gov/site-help/electronic-sub-ref-mat.html.

Desktop Guide: Review Process for 10 CFR 2.206 Petitions
https://www.nrc.gov/about-nrc/regulatory/enforcement/petition.html

NUREG-Series Publications—

NUREG-0750, "Nuclear Regulatory Commission Issuances," published semi-annually: available at http://www.nrc.gov/reading-rm/doc-collections/nuregs/staff/sr0750/.

NUREG/BR-0200, Revision 5, "Public Petition Process," available at http://www.nrc.gov/reading-rm/doc-collections/nuregs/brochures/br0200/.

United States Code

Freedom of Information Act (5 U.S.C. 552).

DH 8.11	REVIEW PROCESS FOR 10 CFR 2.206 PETITIONS	DT-19-01
Volume 8:	Licensee Oversight Programs	
Approved By:	Margaret M. Doane, Executive Director for Operations	
Date Approved:	March 1, 2019	
Cert. Date:	N/A, for the latest version of any NRC directive or handbook, see the online MD Catalog	
Issuing Office:	Office of Nuclear Reactor Regulation Division of Operating Reactor Licensing	
Contact Name:	Perry Buckberg	

EXECUTIVE SUMMARY

Management Directive (MD) 8.11, "Review Process for 10 CFR 2.206 Petitions," is being revised to—

- Clarify the initial screening and acceptance criteria for evaluating petitions,
- Clarify guidance regarding coordination and referral of allegations,
- Clarify and update roles and organizational responsibilities,
- Clarify and add guidance regarding referrals from adjudicatory boards and the Commission,
- Clarify guidance on public meeting and teleconference interactions,
- Clarify guidance for a streamlined director's decision in certain cases,
- Correct the addressee of the periodic 2.206 status report from the Commission to the Director of the Office of Nuclear Reactor Regulation,
- Revise the process to accelerate the PRB initial assessment prior to meeting with the petitioner,
- Add a timeliness goal for issuing the acknowledgment or closure letter,
- Add criteria for holding a petition in abeyance,
- Clarify that the PRB chairperson is the final decision maker for the PRB,
- Add guidance on requests to impose requirements outside of NRC jurisdiction,
- Add the Office of International Programs to the offices responsible for petitions, and
- Relocate detailed procedural staff guidance to "Desktop Guide: Review Process for 10 CFR 2.206 Petitions," to clarify and facilitate future updates, as needed.

For updates or revisions to policies contained in this MD that were issued after the MD was signed, please see the Yellow Announcement to Management Directive index (YA-to-MD index).

TABLE OF CONTENTS

EXHIBIT

I. INTRODUCTION

A. Title 10 of the *Code of Federal Regulations*, Section 2.206

1. Section 2.206 of Title 10 of the *Code of Federal Regulations* (10 CFR 2.206) has been a part of the U.S. Nuclear Regulatory Commission's (NRC's) regulatory framework since the NRC was established in 1975. Section 2.206 permits any person to file a request to institute a proceeding pursuant to Section 2.202 of 10 CFR to modify, suspend, or revoke a license, or for other action as may be proper (hereinafter referred to in this directive as to take enforcement-related action). Such a request is referred to as a 2.206 petition.

2. Section 2.206 requires that a request be submitted in writing, specify the action requested, and set forth the facts that constitute the basis for the request.

3. The NRC staff will not treat general opposition to nuclear power or a general assertion of a safety problem, without supporting facts, as a formal request under 10 CFR 2.206. The staff will treat general requests as allegations or routine correspondence.

4. In addition to receiving petitions as described in 10 CFR 2.206, the Commission or a licensing board may refer issues to the staff for consideration in the 2.206 process.

B. Petitions Containing Allegations of Wrongdoing

1. The NRC defines wrongdoing by NRC licensees or other regulated entities as a willful violation of regulatory requirements (i.e., a violation involving either deliberate misconduct or careless disregard).

2. If a petition alleges wrongdoing on the part of a licensee or other regulated entity, the NRC staff will coordinate with the appropriate office allegation coordinator to enter the petition (or relevant portion thereof) in the allegation program.

3. The Office of the Inspector General (OIG) addresses suspected wrongdoing by NRC employees and contractors such as mismanagement of agency programs that could adversely impact matters related to public health and safety.

4. If the petition contains information of suspected wrongdoing involving an NRC employee, contractor, or vendor, the NRC staff will follow the procedures in Management Directive (MD) 7.4, "Reporting Suspected Wrongdoing and Processing OIG Referrals," for reporting to the OIG.

5. The Director of the Office of Investigations (OI) or the Inspector General (IG), respectively, must approve any mention outside of the NRC of an ongoing OI or OIG investigation.

II. INITIAL STAFF ACTIONS

A. NRC's Receipt of a Petition

1. Process Summary

After the NRC receives a request under 10 CFR 2.206, the Executive Director for Operations (EDO) assigns it to the director of the appropriate office for evaluation and response. After the EDO assigns the petition to the appropriate office, the assigned staff will perform an initial screening of the petition to determine whether it should be entered into the 2.206 process. If the petition is entered into the 2.206 process, a petition review board (PRB) will perform an initial assessment to determine whether it should be accepted for review. If the NRC accepts the petition for review, the official response is the office director's written decision addressing the issues raised in the petition. In that decision, the office director may grant, partially grant, or deny the petitioner's requested action. The NRC provides the petitioner opportunities to address and provide feedback to the PRB. The Commission may, on its own initiative, review the office director's decision within 25 days of the date of the decision, although it will not entertain a request for review of the office director's decision.

2. Assignment of Staff Action and Initial Screening

The assigned staff should perform initial screening of the submittal to determine if the petition, or portions of the petition, should be entered into the 2.206 process. The initial screening criteria are described below:

(a) Issues referred to the staff for consideration as a 2.206 petition by the Commission or a presiding officer in an NRC adjudicatory proceeding will be entered into the 2.206 process as described in Section II.A.2(g) of this handbook.

(b) Petitions may be in the form of requests for an enforcement-related action that may or may not cite 10 CFR 2.206 and may initially be directed to staff other than the EDO. Upon receipt of a written request for an enforcement-related action, regardless of how received, the staff will screen the request to determine if it is within the scope of the 10 CFR 2.206 process.

(c) The staff will promptly review the petition to determine if it requests short-term immediate action (e.g., a request to shut down an operating facility or prevent restart of a facility that is ready to restart) or if an issue raised in the petition may warrant immediate action (even if not requested). See Section III.B.1 of this handbook for more information.

(d) The staff may screen out a request from the 10 CFR 2.206 process and, instead, respond using another appropriate process, such as general correspondence or referral to the allegations process, in the following cases:

(i) Verbal Requests

A verbal request for enforcement-related action under 10 CFR 2.206 (e.g., by telephone or orally in person) will not be considered under the 2.206 petition process. The staff should inform a person who makes a verbal request that the request must be submitted to the NRC in writing using one of the methods described in 10 CFR 2.206. For electronic submissions, "Guidance for Electronic Submissions to the NRC" is available at https://www.nrc.gov/site-help/electronic-sub-ref-mat.html.

(ii) General Assertions and Duplicative Requests for Action under 10 CFR 2.206

The petition is simply (1) a general statement of opposition to licensed activities, nuclear facilities or materials or (2) a general assertion without supporting facts. Examples include conclusory statements without support (e.g., a claim that the quality assurance at a facility is inadequate, with no further explanation), letters submitted to the NRC as a result of mass mailing campaigns, or letters of support for a 10 CFR 2.206 petition that is already under NRC consideration. The staff will not address general assertions with no supporting facts or duplicative requests for action under the 2.206 petition process.

(iii) Allegations

- If the petition alleges wrongdoing (see Section I.B of this handbook), the staff should refer to the allegation program guidance found in MD 8.8, "Management of Allegations" and the Allegation Manual. Referrals to the allegation program should be completed in a timely manner in accordance with MD 8.8.
- The assigned staff should coordinate with the office allegation coordinator and office 2.206 petition coordinator to ensure they reach agreement on any request for action (or portion thereof) that will be referred to the allegation program, including how the submitter will be informed and how the referral will be documented.
- If the staff determines that a petition (or portions thereof) should be referred to the allegation program, those portions of the petition and any correspondence related to the allegation should be handled as prescribed in MD 8.8. In addition, the identity of the petitioner should be protected to the extent practicable with respect to those portions of the petition.

- Once agreement is reached that all or part of a request will be referred to the allegation program, the staff will inform the submitter which parts of the request have been screened out of the 2.206 process, and how the remaining portions will be handled.

- The staff will review any portion of the request that does not involve allegations against the screening criteria in Section II.A.2(d) of this handbook, and will create a public version of the document (with information pertaining to allegations redacted).

- The NRC will redact any information related to allegations contained in the petition from documents sent to the licensee or made available to the public.

(iv) Requests for Non-Public Process or Identity Protection

If a petitioner requests that the petition remain non-public, and/or requests identity protection as part of the process, the staff should explain to the petitioner that the 2.206 process is a public process and, therefore, the petition and petitioner's identity must remain public. The staff should inform any petitioner who does not agree to these terms that the petition will be screened out of the 2.206 process and will be addressed through the appropriate NRC process, such as an allegation or as general correspondence. If the request is transferred to the allegation program, the assigned NRC staff will coordinate with the office allegation coordinator, consistent with MD 8.8.

(v) Requests That Would Not Reasonably Lead to an Enforcement Action

NRC regulations state that a 2.206 petition is a request "to institute a proceeding pursuant to 10 CFR 2.202 to modify, suspend, or revoke a license, or for any other action as may be proper." The regulations also require that the request "specify the action requested and set forth the facts that constitute the basis for the request."

- A petition should be screened out if it does not request a specific enforcement-related action (e.g., issuing an order modifying, suspending, or revoking a license pursuant to 10 CFR 2.202, issuing a notice of violation pursuant to 10 CFR 2.201, etc.) and does not identify a specific safety or security concern (e.g., a technical deficiency or potential violation). A petition must provide information that could reasonably lead the NRC to take an enforcement action (not necessarily the action requested).

- A petition that identifies a valid safety or security concern will not be screened out solely because the action requested is inappropriate for the circumstances.
- A petition that does not request a specific enforcement-related action should be evaluated to determine if it contains an implied request for action. If a petition does not contain an explicit or implied request for enforcement-related action, the request should be screened out of the 2.206 process and be considered for referral to an appropriate NRC process (e.g. allegations, rulemaking, or general correspondence).

(vi) Requests to Impose a Requirement that is Outside of NRC Jurisdiction

A request to impose a requirement that is outside the jurisdiction of the Commission (e.g., a state or local ordinance or a requirement of another federal agency) will not be considered under the 2.206 process, but may be referred to the appropriate regulatory authority.

(vii) Requests for Rulemaking

A petition that alleges deficiencies in existing NRC rules, and/or requests changes to existing NRC rules, will not be considered under the 2.206 process, but may be referred to the appropriate rulemaking branch for consideration as a petition for rulemaking under 10 CFR 2.802. The petition manager will consult with the appropriate rulemaking branch within the NRC, and will incorporate the rulemaking branch's input into the NRC's response to the petitioner.

(viii) Requests for Information

If a petition contains a request for public records regarding NRC licensed activities, nuclear facilities or materials licensees, that request will not be considered under the 2.206 process. In such cases, the petitioner should be referred to the NRC Freedom of Information Act (FOIA) Guide. The FOIA generally provides any person the right to obtain access to Federal agency records.

(ix) Issue(s) Under Review in an Adjudicatory Proceeding

If the issue(s) raised in a petition (or portions thereof) are the subject of a proffered or admitted contention in an ongoing NRC adjudicatory proceeding regarding the same licensee and facility, those issues generally will not be considered in the 2.206 process (regardless of whether the 2.206 petitioner proffered the contention or is a party to the proceeding).

(e) Notwithstanding the screen-out criteria above, the staff, upon its own determination, may consider an issue for immediate action and/or inclusion in the 2.206 process.

(f) For requests that are screened out, the staff should inform the submitter of the reasons why, referring back to the screen-out criteria above, and explain that the concern(s) raised will be transferred to another process (e.g., petition for rulemaking, or general correspondence). The communication of the staff's decision to screen out a request and refer it to another process should be documented as an official agency record (e.g., e-mail added in ADAMS, or record of a phone call).

(g) A request for an enforcement-related action that is not screened out under Section II.A.2 will be entered into the 2.206 petition process and evaluated for acceptance as described in Section III.C of this handbook.

B. Petition Manager Initial Action

1. The petition manager will promptly review the petition to determine if it requests short-term immediate action (e.g., a request to shut down an operating facility or prevent restart of a facility that is ready to restart) or if an issue raised in the petition may warrant immediate action (even if not requested). See Section III.B.1 of this handbook for more information on immediate requests.

2. Before the petition is released to the public and before the PRB meeting, the petition manager will informally inform the petitioner the petition was received and, because the 2.206 petition process is a public process, the petition and all the information in it, including the petitioner's identity, will be made public.

3. After the initial contact with the petitioner, the petition manager will promptly advise relevant licensee(s) of the petition, and send the appropriate licensee(s) a copy of the petition for information.

4. See the "Desktop Guide: Review Process for 10 CFR 2.206 Petitions," for further information on petition manager actions. The Desktop Guide is available on the NRC public webpage).

III. PETITION REVIEW BOARD (PRB)

A. Petition Review Board Composition

The PRB consists of—

1. A PRB chairperson (generally a Senior Executive Service manager).

2. The office 2.206 petition coordinator.

3. A 2.206 petition manager.

4. Cognizant management and staff, as necessary.
5. A cognizant regional representative (a regional branch chief or higher if there is a concern involving a potential violation).
6. A representative from OI, if recommended by the petition manager.
7. A representative from the Office of Enforcement (OE). The OE representative should address both the enforcement and allegation programs and inform the PRB if the petition involves an issue that is already in, or was previously addressed in, the allegation or enforcement programs.
8. The petition manager may also recommend that the office enforcement coordinator be included in the PRB.
9. A representative from the Office of the General Counsel, as necessary.

B. Schedule for PRB Meeting

1. If the petition requests immediate action or the petition manager determines that immediate action may be necessary, the petition manager will convene an initial PRB meeting as soon as possible to decide whether immediate action is warranted. The petition manager may hold an in-person meeting of the PRB or use other means (e-mail, teleconference) to obtain the PRB's recommendation on immediate actions. In such cases, a subsequent PRB meeting (see Section III.D of this handbook) will be held to evaluate the petition for acceptance. In extremely urgent cases that do not enable formation of a PRB, the petition manager will consult with office management to ensure the petition is appropriately addressed. Immediate actions are approved or denied by the assigned office director.
2. After addressing any requests for immediate action (see Section III.B.1 above), the assigned office will convene a PRB meeting to evaluate the petition for acceptance. The PRB meeting should be held as quickly as possible, but no later than 3 weeks after EDO assignment of the petition. See Section IV.B of this handbook for more information on establishing a schedule for the PRB's review.

C. Criteria for Petition Evaluation

The staff will use the criteria in this section to determine whether to accept a petition for review, whether to consolidate two or more petitions, and whether to hold a petition in abeyance.

1. Criteria for Accepting Petitions Under 10 CFR 2.206

   The staff will accept a petition, or a portion of the petition, for review under 10 CFR 2.206 if the request meets the criteria in Section III.C.1(a) and (b) below:

(a) The petition specifies the facts that constitute the basis for taking the requested action, and those facts are sufficient to provide support for the requested action. The petitioner must provide more than a bare assertion that the NRC should take action. The supporting facts must be sufficient to warrant further inquiry.

(b) The petition falls within one of the following categories:

(i) The issues raised by the petitioner have not previously been the subject of a facility-specific or generic NRC staff review, or

(ii) The issues raised have previously been the subject of a facility-specific or generic NRC staff review, and at least one of the following circumstances applies:

- The prior review did not resolve the issues raised by the petitioner, or
- The resolution of the issues in the prior review does not apply to the facts provided by the petitioner to support the requested action, or
- The petition provides significant new information that the staff did not consider in the prior review.

(c) For the criterion in Section III.C.1(b)(ii) above:

(i) If the prior review occurred in the allegation process, the petition (or portion thereof) will not be accepted in the 2.206 process. Rather, the staff's prior conclusion will be shared publicly without reference to the related allegation.

(ii) In other cases involving prior reviews, the staff should determine, in its technical judgment, whether or not the listed circumstances in Section III.C.1(b)(ii) apply. In most cases, if the staff determines that an issue has been resolved, the staff should identify its supporting documentation.

(d) If the petition raises multiple issues, the staff should accept the petition only with respect to those issues that satisfy the criteria in Section III.C.1(a) and (b) above.

2. Criteria for Consolidating Petitions

Generally, all requests submitted by different individuals will be treated and evaluated separately. When two or more petitions request action against the same licensee, specify essentially the same bases, provide adequate supporting information, and are submitted at about the same time, the PRB must weigh the benefit of consolidating the petitions against the potential for minimizing the importance of any single petition. The PRB will recommend whether consolidation is or is not appropriate, and the assigned office director or deputy office director will make the final determination.

3. Criteria for Holding a Petition in Abeyance

   If a petition meets the acceptance criteria in Section III.C.1 of this handbook, there may be circumstances in which it would be appropriate to hold the petition in abeyance pending the outcome of a related staff review outside of the 2.206 process.

   (a) The PRB may hold a petition in abeyance if—

      (i) The issues raised in the petition are the subject of ongoing or imminent review,

      (ii) The review is not expected to be completed in the near future, and

      (iii) The staff needs the results of the review in order to reach an informed decision on the issues raised in the petition.

   (b) If the petition raises multiple issues, the PRB should hold in abeyance only those portions of the petition that meet the criteria in Section III.C.3(a) above.

   (c) The staff should not hold a petition in abeyance solely to allow a petitioner to develop additional supporting information not provided with the original petition.

   (d) When the PRB decides to hold all or part of a petition in abeyance—

      (i) The PRB chairperson will ensure that the office director, or designee, is informed of the PRB's decision and concurs with the decision.

      (ii) The petition manager will then inform the petitioner of the PRB decision and its basis.

      (iii) The petition manager will also inform the petitioner when the PRB expects to resume its assessment of the 2.206 petition.

      (iv) If a petition is held in abeyance, the petition manager will notify the petitioner by telephone and/or e-mail that status updates will occur at least every 120 days (unless another time period is agreed upon with the petitioner) as described in Section IV.C of this handbook.

      (v) When the staff completes its review of the related issue, the petition manager will notify the petitioner that the petition is no longer being held in abeyance and the PRB is resuming its review.

D. PRB Initial Assessment

1. The PRB ensures that the staff follows an appropriate process in evaluating a petition. The PRB—

   (a) Determines whether the petitioner's request meets the criteria for accepting petitions for review (see Section III.C.1 of this handbook).

(b) Determines whether there is a need for immediate action (whether requested or not).

(c) Establishes a schedule for responding to the petitioner in a timely manner (see Section IV.B of this handbook for guidance regarding schedules).

(d) Determines whether the petition should be consolidated with another petition.

(e) Confirms whether any referrals to the allegation program or OIG made during initial screening are appropriate.

(f) Determines whether the licensee should be asked to respond to the petition.

(g) Addresses the possibility of issuing a streamlined director's decision concurrently with the acknowledgment letter for cases where the basis of the petition is well known to the NRC staff and existing regulatory framework is in place to address the concerns raised. See Section III.G.2(f) of this handbook for information on when a streamlined response could be appropriate.

2. The PRB meetings to consider immediate actions, evaluate the petition against the acceptance criteria, or to review the petition are closed to the public and separate from the PRB meetings with the petitioner and the licensee described in Section III.F of this handbook.

(a) At the meeting, the petition manager briefs the PRB on the petitioner's request(s), any background information, the need for an independent technical review, and a proposed plan for resolution, including target completion dates.

(b) The petition manager, with the assistance of the office 2.206 petition coordinator, ensures appropriate documentation of all PRB recommendations in the summary of the PRB meeting.

E. Informing the Petitioner of the Results of the Initial PRB Assessment

1. After the PRB performs the initial assessment of the petition against the evaluation criteria in Section III.C of this handbook, and before meeting with the petitioner, the PRB chairperson will inform the office director, or designee, of the results of the PRB's initial assessment.

2. The petition manager will then inform the petitioner of the following:

(a) Whether or not the petition, as submitted, meets the criteria for acceptance in Section II.C.1 of this handbook.

(b) The disposition of any request for immediate action.

(c) If the petition is accepted for review, the process the PRB will follow to review the petition.

(d) The opportunity to meet with the PRB to discuss the initial assessment, as described in Section III.F of this handbook.

(e) If the petitioner chooses to meet with the PRB, any questions or comments on the petition that the PRB would like the petitioner to address.

3. If the staff plans to take an action that is contrary to an immediate action requested in the petition before issuing either the closure letter or acknowledgment letter, the petition manager should informally notify the petitioner promptly by telephone and/or e-mail of the pending staff action. Reasons for the staff's action will be documented in the closure or acknowledgment letter.

4. The petitioner will not be advised of an ongoing investigation of wrongdoing being conducted by OI, but should be informed if the petition contained an assertion of wrongdoing that is being referred to the allegation program for possible investigation.

F. Meeting With the Petitioner

1. After informing the petitioner of the results of the PRB's initial assessment, the petition manager will offer the petitioner an opportunity for a public meeting with the PRB to clarify or supplement the petition based on the results of the PRB's initial assessment. The meeting between the PRB and the petitioner, if accepted, will be held as a public meeting, either in-person at NRC headquarters in Rockville, Maryland, or by another agreed-upon arrangement (e.g., public teleconference or virtual public meeting). This public meeting should be scheduled so as not to adversely affect the established petition review schedule.

(a) If the petitioner chooses to address the PRB by teleconference, the petition manager will establish a mutually agreeable time and date and arrange to conduct the teleconference on a moderated and recorded bridge line. The petition manager will arrange for transcription service and the transcript will become a supplement to the petition.

(b) If the petitioner accepts the offered meeting with the PRB, the petition manager will establish a mutually agreeable time and date for the meeting with the petitioner. The petition manager will follow the public notice period and other provisions of MD 3.5, "Attendance at NRC Staff-Sponsored Meetings." The meeting should be referred to as a meeting between the NRC staff, the petitioner, and the licensee (unless the licensee chooses not to participate). The meeting will be available through a moderated and recorded bridge line and a transcript will be created and distributed to the same distribution list as the original petition.

2. This meeting with the PRB, if held, is an opportunity for the petitioner to provide any relevant additional explanation and support for the request in light of the PRB's initial assessment. The PRB will consider the petitioner's statements made at the meeting,

along with the original petition, in making its final recommendation on whether to accept the petition according to the criteria in Section III.C.1 of this handbook.

3. If the petitioner presents significant new information to the NRC staff that is unrelated to the concerns raised in the petition, the PRB may determine that the new information constitutes a new petition.

4. The petition manager will invite the licensee to participate in the meeting with the petitioner to ensure that the licensee understands the concerns about its facility or activities.

5. During the meeting with the petitioner, the PRB members may ask questions of the petitioner or the licensee to clarify their understanding of the issues raised in the petition. After the petitioner's presentation, the PRB will give the licensee an opportunity to ask the PRB members questions related to the issues raised in the petition. Also, the PRB will give the petitioner and the licensee an opportunity to ask the PRB questions related to the process for evaluating and reviewing 2.206 petitions. Although the intent is that the PRB members would respond to such questions, the licensee or petitioner may also voluntarily respond. If detailed information is needed from the licensee, the PRB should ask the licensee to provide a voluntary response as discussed in Section IV.A.1 of this handbook.

6. The petition manager will ensure that all NRC staff at the meeting are aware of the need to protect sensitive information from disclosure.

7. The petitioner may request that a reasonable number of associates be permitted to assist in addressing the PRB at the meeting. The petition manager will—

   (a) Discuss this request with the petitioner,

   (b) Determine the number of speakers, and

   (c) Allot a reasonable amount of time for the presentation so that the staff can acquire the information needed for its review in an efficient manner.

8. Prior to concluding the meeting, the petition manager will request feedback from attendees on the 2.206 review process. Such feedback may be provided during the meeting or after the meeting (using the public meeting feedback survey or by directly contacting the petition manager). Staff who receive feedback should discuss the input received with their office 2.206 petition coordinator and their management as appropriate.

9. The petition manager will review the meeting transcript, and where necessary, edit it to ensure it accurately reflects what was said in the meeting. Corrections are only necessary for errors that affect the meaning of the text of the transcript. The petition manager is not expected to correct inconsequential errors.

10. After editing, the petition manager will ensure that the transcript receives the same distribution (petitioner, licensee, publicly available in ADAMS, etc.) as the original petition.

11. After the meeting with the petitioner, the PRB will consider the supplemental information presented during the meeting together with the original petition in making its final recommendation on whether to accept the petition for review. Before issuing either an acknowledgment or closure letter, the PRB chairperson will ensure that the office director, or deputy office director, is informed of the PRB's recommendations (including a recommendation to issue a partial or streamlined director's decision) and concurs with the recommendations.

G. Response to the Petitioner

1. The petition manager will promptly notify the petitioner by e-mail about NRC staff decisions regarding immediate action requests. Such notifications may occur before the PRB finalizes its recommendation on whether to accept the petition for review.

2. After the PRB finalizes its recommendations on whether to accept the petition for review, the petition manager will notify the petitioner of the PRB's determination by telephone and/or e-mail. If the petition is accepted, the petition manager will inform the petitioner of how the review will proceed. The PRB's recommendations will be documented in either a closure letter (which documents the reasons why the petition was not accepted for review) or an acknowledgment letter (if the petition is accepted for review). The closure letter or acknowledgment letter will address any supplemental information provided by the petitioner, any comments the petitioner made concerning the initial PRB assessment, and the NRC staff's response to those comments. Section IV.B, "Schedule," of this handbook describes planning the schedule specifying the goal for the acknowledgment or closure letter to be issued within 90 days of the EDO assigning the petition.

3. Requests That Do Not Meet the Criteria for Acceptance

   (a) If the PRB, with office-level management concurrence, determines that the petition does not meet the criteria for acceptance as a 10 CFR 2.206 petition, the petition manager then prepares a closure letter that—

      (i) Explains why the request was not accepted for review under 10 CFR 2.206, referring back to the Criteria for Petition Evaluation in Section III.C of this handbook,

      (ii) Acknowledges the petitioner's efforts in bringing issues to the staff's attention,

      (iii) If applicable, explains the staff's response to the immediate action requested and the basis for that response,

      (iv) Notifies the petitioner whether the request is being referred to another NRC program for action, and

(v) Responds, to the extent possible at that time, to the issues in the petitioner's request and identifies supporting documents if applicable.

(b) The assigned organization is responsible for ensuring the appropriate concurrence and distribution for the closure letter. At a minimum, each PRB member and the office director concurs on the closure letter. The PRB chairperson signs the closure letter.

4. Requests That Meet the Criteria for Acceptance

(a) If the PRB finds that the petition meets the criteria for acceptance as a 10 CFR 2.206 petition, the petition manager prepares an acknowledgment letter and associated *Federal Register* notice of receipt. See the "Desktop Guide: Review Process for 10 CFR 2.206 Petitions," available on the NRC public webpage at, for more details.

(b) The letter should acknowledge the petitioner's efforts in bringing issues to the staff's attention.

(c) If the petition contains a request for immediate action by the NRC, the acknowledgment letter will explain the staff's response to the immediate action requested and the basis for that response.

(d) The petition manager ensures that references MD 8.11 and NUREG/BR-0200, Revision 5, "Public Petition Process," are included with the acknowledgment letter. A copy of the acknowledgment letter must be sent to the appropriate licensee and the docket service list(s). See the "Desktop Guide: Review Process for 10 CFR 2.206 Petitions," available on the NRC public Web page.

(e) The assigned organization is responsible for ensuring the appropriate concurrence and distribution for the acknowledgment letter. At a minimum, each PRB member concurs on the acknowledgment letter. The office director signs the acknowledgment letter.

(f) Streamlined Director's Decisions

(i) If the petition meets the criteria for acceptance but raises issues that the staff has evaluated and is prepared to issue a decision on, the staff may respond immediately to the petition by issuing a streamlined director's decision. Issuing a streamlined director's decision allows the NRC to move forward with an imminent decision or action that appropriately considers the information in the petition and avoids unnecessary duplication of NRC resources by the PRB addressing the same issue. For example, a streamlined director's decision may be appropriate in a case where a petition's supporting information consists almost entirely of NRC-generated information (e.g., inspection reports, generic letters) or information well known to the NRC (e.g., news reports, licensee event reports). In these cases, a proposed director's

decision would not be issued, and the acknowledgment letter would be accompanied by the final director's decision.

(ii) Before issuing a streamlined director's decision, the PRB will consider the need to contact the petitioner to determine if the petitioner possesses information relevant to the bases for the decision that is beyond what is currently available to the NRC. In most cases, a streamlined director's decision would be issued without this additional interaction with the petitioner, and the petitioner can provide feedback after issuance.

(iii) The petition manager will inform the petitioner of plans to issue a streamlined director's decision.

H. Providing Documents to the Petitioner

1. If the PRB determines that the 2.206 petition will be accepted for review, then the petition manager will—

   (a) Add the petitioner to the service list(s) for the topic (if one exists). If a listserv is used, the petition manager will inform the petitioner how to join the listserv to receive electronic versions of the NRC's publicly available outgoing correspondence.

   (b) Send copies electronically of any future correspondence from the licensee related to the petition to the petitioner, with due regard for proprietary, safeguards, and other sensitive information in accordance with established agency policies and procedures.

   (c) Ensure that the petitioner is placed on distribution for other NRC correspondence relating to the issues raised in the petition, to the extent that the petition manager is aware of these documents, including relevant NRC generic communications (i.e., generic letters, regulatory issue summaries, information notices, or bulletins) that are issued while the NRC considers the petition. The petition manager will inform the petitioner how to join the listserv to receive electronic versions of publicly available NRC generic communications.

2. These three actions will remain in effect until 90 days after the director's decision is issued if the petitioner desires it.

I. Supplements to the Petition

A petitioner will occasionally submit a written supplement to a petition.

1. When a supplement is provided, the petition manager will promptly review the supplement to determine whether or not it contains sensitive information, which must be handled according to appropriate information security policies and procedures.

2. The petition manager will then include the supplement in the ongoing acceptance review (if the supplement is received before the PRB makes its final determination) or petition review (if the petition has been accepted) by taking appropriate actions listed in Section II.B of this handbook. The petition manager will ensure that the supplement receives the same distribution as the petition and will forward a copy of the supplement to the PRB members. The PRB members will review the supplement and determine whether they need to meet formally to discuss it and, if so, whether or not to offer the petitioner an opportunity to discuss the supplement with the PRB. In deciding whether an additional PRB meeting is needed, the PRB members will consider the safety significance and complexity of the information in the supplement. Clarification of previous information will generally not require an additional PRB meeting.

3. When a supplement is received, the petition manager will inform the petitioner of the PRB's schedule and advise the petitioner that additional supplements could delay the evaluation of the petition for acceptance or the review of a petition that has been accepted. Supplements will be considered to the extent practical taking into account the petition review schedule. Any impacts to the petition review schedule should be kept to a minimum.

4. The PRB will review supplements for additional relevant explanation or clarification of the issues raised in the original petition or additional relevant facts supporting the petitioner's view of the issues. To the extent that supplemental information provided by the petitioner raises new issues, requests additional enforcement-related actions, or otherwise expands the scope of the original petition, the PRB may consider such information as amending the petition and decline to consider the supplemental information in the petition review process. If the petitioner presents significant new information to the NRC staff, the PRB may determine that the supplement constitutes a new petition that will be treated separately from the initial petition.

5. After receiving a supplement, the PRB will then determine whether—

   (a) There is a need for any immediate actions based on the supplemental information (whether requested or not).

   (b) The supplement should be consolidated with the existing petition.

   (c) The petition, as supplemented, meets the criteria for acceptance in Section II.C.1 of this handbook (if the petition has not already been accepted for review).

   (d) To issue a partial director's decision.

   (e) To revise the review schedule for the petition based on the supplement (see Section IV, "Petition Review Activities," of this handbook for guidance regarding schedules).

(f) To send a letter acknowledging receipt of the supplement. A letter should be sent if the supplement provides significant new information, causes the staff to reconsider a previous determination, or requires a schedule change beyond the original 120-day goal.

(g) To offer the petitioner a meeting or teleconference with the PRB to discuss its recommendations with respect to the supplement. See Section III.F of this handbook for information on this type of meeting or teleconference.

6. For supplements received after an acknowledgment letter has been issued, the staff may determine that the schedule for the petition must be extended beyond the original goal as a result of the supplement. In this case, the assigned office should send an acknowledgment letter to the petitioner, reset the clock to the date of the new acknowledgment letter, and inform the OEDO.

7. If the PRB determines that the supplement will be treated as a new petition (i.e., not consolidated with the existing petition), the assigned office must contact OEDO for a new tracking number.

IV. PETITION REVIEW ACTIVITIES

This section describes the activities that take place after a petition has been accepted for review.

A. Reviewing the Petition

1. Request for Licensee Input

(a) If appropriate, the petition manager will request the licensee to provide a voluntary response to the NRC on the issues specified in the petition, usually within 30 days. This staff request usually will be made in writing. The petition manager will advise the licensee that the NRC will make the licensee’s response publicly available and will provide a copy of the response to the petitioner. The licensee may also voluntarily submit information related to the petition, even if the NRC staff has not requested this information.

(b) Unless necessary for the NRC’s proper evaluation of the petition, the licensee should avoid using proprietary or personal privacy information that requires protection from public disclosure. If this information is necessary to completely respond to the petition, the petition manager ensures the information is protected in accordance with 10 CFR 2.390, “Public Inspections, Exemptions, Requests for Withholding.”

2. Technical Review Meeting With the Petitioner

The staff will hold a technical review meeting with the petitioner whenever it believes that a meeting (whether requested by the petitioner, the licensee, or the staff) would be beneficial to the staff’s review of the petition. Meeting guidance is provided in

MD 3.5. The petition manager will ensure that the meeting does not compromise the protection of sensitive information. A meeting will not be held simply because the petitioner claims to have additional information and will not present it in any other forum.

3. Additional PRB Meetings

   Additional PRB meetings may be scheduled for complex issues. Additional meetings also may be appropriate if the petition manager finds that significant changes must be made to the original plan for the resolution of the petition.

4. Conduct of PRB Meetings

   The PRB chairperson makes the final decisions regarding recommendations proposed during the PRB meeting and provides final approval for requested actions. The petition manager prepares for and documents decisions made during the PRB meeting.

B. Schedule

Planning the Schedule

1. The first goal is to issue the acknowledgment or closure letter within 90 days of the OEDO assigning the petition.
2. The second goal is to issue the proposed director's decision for comment within 120 days after issuing the acknowledgment letter. The proposed director's decision for uncomplicated petitions should be issued in less than 120 days.
3. The third goal is to issue the final director's decision within 45 days of the end of the comment period for the proposed director's decision. The actual schedule should be shorter if the number and complexity of the comments allow.

C. Keeping the Petitioner Informed

The petition manager ensures that the petitioner is notified at least every 60 days of the status of the petition, or more frequently if a significant action occurs. In cases where a petition is being held in abeyance, the petition manager ensures that the petitioner is notified at least every 120 days (or other timeframe agreed upon with the petitioner) and when the staff is ready to resume its review of the petition. The petition manager provides updates to the petitioner by telephone and/or e-mail. The petition manager should speak directly to the petitioner if reasonably possible. The petition manager must monitor the status of the petition so that reasonable detail can be provided. However, the update to the petitioner will not identify or discuss—

1. An ongoing OI or OIG investigation, unless approved by the Director of OI or the IG;
2. The referral of the matter to the Department of Justice (DOJ); or

3. Enforcement action under consideration.

D. Updating NRC Management and the Public

1. On a quarterly basis, the Division of Operating Reactor Licensing, NRR, will issue a status report of 2.206 petitions to the Director of NRR. The agency 2.206 petition coordinator also ensures the status report is added to ADAMS and made publicly available.

2. The NRC Web site provides petitions filed, director's decisions issued, quarterly status reports, and other related information, available at https://www.nrc.gov/about-nrc/regulatory/enforcement/petition.html.

V. THE DIRECTOR'S DECISION

A director's decision is the official agency response to a 2.206 petition that is accepted for review. The director's decision may grant, partially grant, or deny the action requested by the petitioner. In most cases, the staff prepares a proposed director's decision, which is distributed to the petitioner and licensee for comment. After receiving any comments, the staff revises the director's decision as appropriate. The director's decision is then issued and a notice of issuance is subsequently published in the *Federal Register*.

A. Content and Format

1. The petition manager prepares a proposed director's decision on the petition for the office director's consideration. The petition manager also prepares letters to the petitioner and the licensee requesting comment on the proposed director's decision.

2. If the staff issues a streamlined director's decision, the steps related to a proposed director's decision may be omitted; see Section III.G.2(f) of this handbook for more information.

3. The proposed director's decision will clearly describe the issues raised by the petitioner, provide a discussion of the safety significance of the issues, and clearly explain the staff's disposition for each issue. If a partial director's decision was issued previously, the final director's decision will refer to, but does not have to repeat the content of, the partial director's decision.

B. Granting the Petition

The NRC may grant a petition for enforcement-related action, either in whole or in part, and it also may take other action to address the concerns raised by the petitioner. Once the staff has determined that a petition will be granted, in whole or in part, the petition manager will prepare a "Director's Decision under 10 CFR 2.206" for the office director's signature. The decision will explain the bases upon which the petition has been granted and identify the actions that the NRC staff has taken, or will take, to grant all or that portion of the petition. The decision also should describe any actions the licensee took

voluntarily that address aspects of the petition. A petition is characterized as being granted in part when the NRC grants only some of the actions requested and/or takes actions other than those requested to address the underlying problem. If the petition is granted in full, the director's decision will explain the bases for granting the petition and state that the NRC's action resulting from the director's decision is outlined in the NRC's order or other appropriate communication. If the petition is granted in part, the director's decision will clearly indicate the portions of the petition that are being denied and the staff's bases for the denial. When granting a petition, either in whole or in part, the PRB should consider guidance and policy in MD 8.4, "Management of Facility-Specific Backfitting and Information Collection."

C. Denying the Petition

When the staff has determined that a petition will be denied, the petition manager will prepare a "Director's Decision under 10 CFR 2.206" for the office director's signature. The decision will explain the bases for the denial and discuss all matters raised by the petitioner in support of the request.

D. Final Versus Partial Director's Decision

1. If all of the issues in the petition can be resolved together in a reasonable amount of time, then the staff will issue one director's decision addressing all of the issues. The staff will consider preparing a partial director's decision when some of the issues associated with the 2.206 petition are resolved in advance of other issues and if significant schedule delays are anticipated before resolution of the entire petition.

2. The format, content, and method of processing a partial director's decision are the same as that of a proposed director's decision and an accompanying *Federal Register* notice of issuance would still be prepared. However, the partial director's decision should clearly indicate those portions of the petition that remain open, explain the reasons for the delay to the extent practical, and provide the staff's schedule for the final director's decision.

3. Once a partial director's decision has been issued, the petition manager will prepare an extension request to extend the due date to support the resolution of any remaining issues. After completing its review of the remaining issues, the staff will issue a final director's decision addressing those issues. The final director's decision will refer to, but does not have to repeat the content of, the partial director's decision.

E. Issuing the Proposed Director's Decision for Comment

1. After the assigned office director has concurred on the transmittal letters and the proposed director's decision, the assigned division director signs the transmittal letters. The petition manager will issue letters to the petitioner and the licensee

requesting comments on the enclosed, fully concurred on but unsigned, proposed director's decision.

2. The intent of this step is to give the petitioner and the licensee an opportunity to share any concerns they may have with the decision. The letters will request comments within a set period of time, typically 2 weeks. The amount of time allowed for comments may be adjusted depending on circumstances. For example, for very complex technical issues, it may be appropriate to allow more time for the petitioner and licensee to develop their comments.

F. Comment Disposition – Proposed Director's Decision

1. After the comment period closes on the proposed director's decision, the assigned office will review the comments received and provide the schedule to issue the director's decision to the agency 2.206 petition coordinator. The petition manager will evaluate any comments received on the proposed decision, obtaining the assistance of the technical staff, as appropriate. Although the staff only requests comments from the petitioner and the licensee, comments from other sources (e.g., other members of the public) may be received. These additional comments should be addressed in the same manner as the comments from the petitioner and licensee. A copy of the comments received and the associated staff responses will be included in the director's decision. An attachment to the decision will generally be used for this purpose.

2. If no comments are received on the proposed decision, the petition manager will include in the director's decision a reference to the letters that requested comments and a statement that no comments were received.

3. If the comments from the petitioner include new information, the PRB will reconvene to determine whether to treat the new information as part of the current petition or to treat it as a new petition which would be screened as described in Section II.A.2 of this handbook.

G. Issuing the Director's Decision

1. The petition manager prepares a transmittal letter to the petitioner and the director's decision (or partial director's decision) to be signed by the office director. In addition, the petition manager prepares a *Federal Register* notice of issuance.

2. If the director's decision grants the issuance of an order, the order will be issued prior to, or concurrent with, issuing the director's decision. The petition manager will include a copy of the order as an enclosure to the transmittal letter to the petitioner.

3. The assigned office is responsible for ensuring the appropriate concurrence and distribution on the transmittal letter to the petitioner.

4. Before providing a director's decision to the office director for signature, the assigned office will contact the agency 2.206 petition coordinator for a director's decision number.

5. The assigned office director will sign the director's decision and the transmittal letter to the petitioner.

6. When the director's decision has been signed, the petition manager will ensure that the agency 2.206 petition coordinator is immediately informed. On the day the director's decision is signed, the agency 2.206 coordinator is expected to inform the Office of the Secretary (SECY) that the director's decision has been issued.

7. The petition manager will promptly inform the petitioner that the director's decision has been signed and will send a courtesy copy of the signed director's decision, electronically if possible, to the petitioner.

8. Occasionally, a petitioner may submit comments on a final decision after it is issued. In this case, the petition manager should ensure that the PRB reviews the comments provided and that an appropriate response is provided within a reasonable amount of time. If the petitioner provides new information in the comments, the PRB should determine whether the decision should be revised or if the information should be treated as a new petition. The petition manager should ensure that the comments and any staff response are added to the ADAMS records associated with the final decision. Any staff receiving feedback should ensure that the respective office 2.206 petition coordinator and management are aware of the feedback to facilitate identification of areas for process improvement.

9. The "Desktop Guide: Review Process for 2.206 Petitions," is available on the NRC public Web page for more specific procedural details.

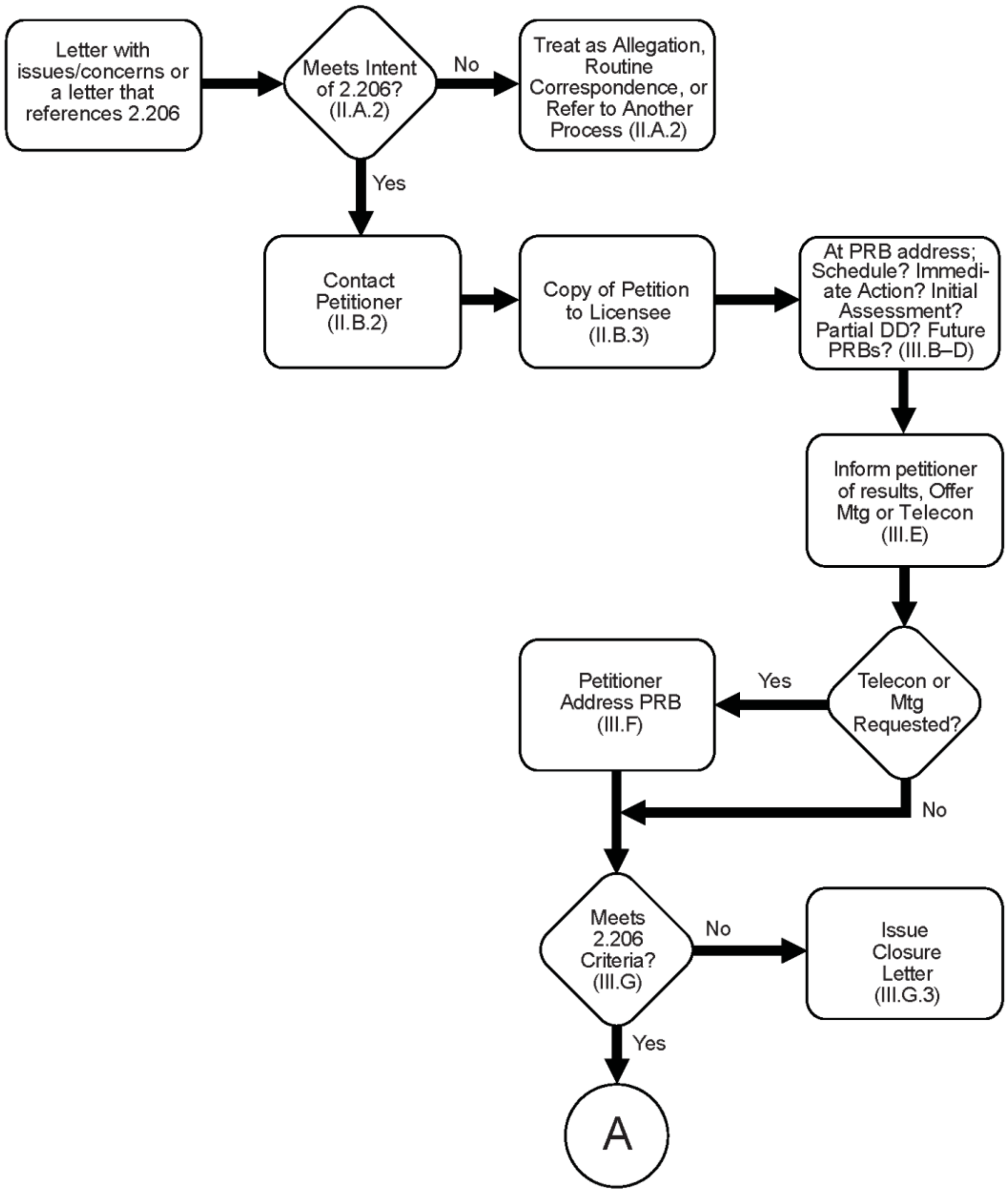
H. Coordination with SECY

1. The agency 2.206 petition coordinator is responsible for requesting a director's decision number from SECY, and for notifying SECY of the issuance of a director's decision on the day the decision is signed. On the day of signature, the staff should keep the agency 2.206 petitioner coordinator informed.

2. When the agency 2.206 petition coordinator provides SECY with the ADAMS accession number of the signed director's decision and the package accession number, SECY will inform the Commission of the availability of the decision. If the director's decision denies the requested action in whole or in part, the Commission, at its discretion, may decide to review the director's decision within 25 days of the date of the decision and, as a result of its review, may direct the staff to take action other than that described in the director's decision. If the Commission does not act on the director's decision within 25 days or decide to extend its review time, the director's decision becomes the final agency action on the petition, and SECY will

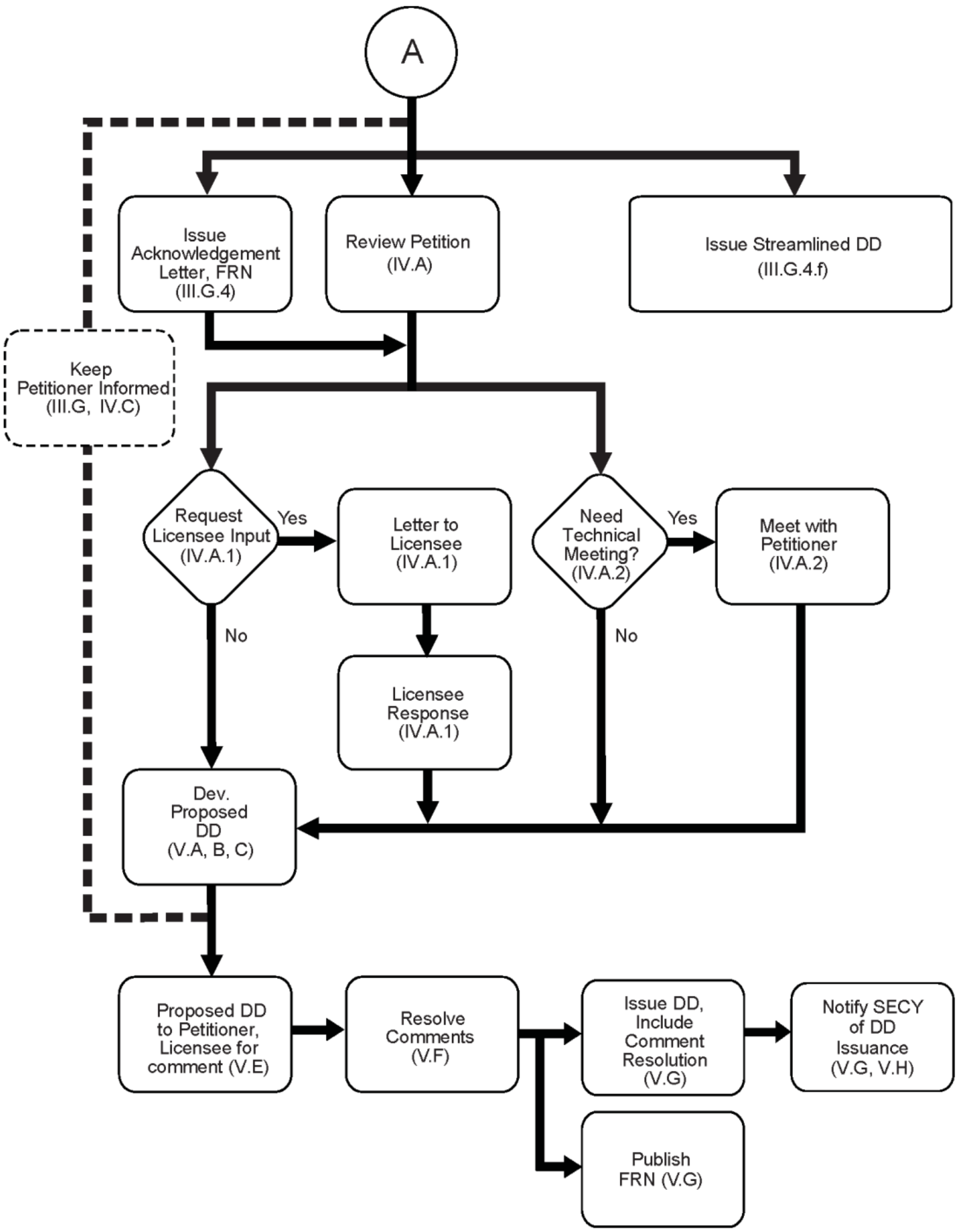
inform the petitioner by letter that the Commission has taken no further action on those portions of the petition addressed in the director's decision.

EXHIBIT Simplified 2.206 Process Flow Chart (1 of 2)



1. Parenthetical Information is associated Handbook paragraph number

EXHIBIT Simplified 2.206 Process Flow Chart (2 of 2)



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)

<u>Operations of an ISFSI (60855)</u>

- 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 2×10^{-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. Pressure Testing

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. <u>Records</u>

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 <u>Conclusions</u>

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;
SONGS ISFSI – DATED AUGUST 24, 2018

DISTRIBUTION:
K. Kennedy, ORA
S. Morris, ORA
T. Pruett, DNMS
L. Howell, DNMS
J. Katanic, FCDB
L. Brookhart, FCDB
E. Simpson, FCDB
W. Allen, SFST
V. Dricks, ORA
A. Moreno, CAO
J. Weil, CAO
C. Cook, OEDO
P. Silva, NMSS
W. Maier, ORA
S. Burnell, OPA
R4_FCDB

cc:

Mr. Al Bates, Regulatory Affairs Manager
Southern California Edison Company
San Onofre Nuclear Generating Station
P.O. Box 128
San Clemente, CA 92674-0128

Mr. Gonzalo Perez, Branch Chief
Radiologic Health Branch
Division of Food, Drug, & Radiation Safety
CA Dept. of Health Services
P.O. Box 997414, MS 7610
Sacramento, CA 95899-7414

Mr. Lou Bosch, Plant Manager
Southern California Edison Company
San Onofre Nuclear Generating Station
P.O. Box 128
San Clemente, CA 92674-0128

Dr. Robert B. Weisenmiller, Chair
California Energy Commission
1516 Ninth Street (MS 34)
Sacramento, CA 95814

Mr. W. Matthews III, Esquire
Southern California Edison Company
Law Department
2244 Walnut Grove Avenue
Rosemead, CA 91770

Mr. Mark Morgan, Regulatory Affairs
Southern California Edison Company
San Onofre Nuclear Generating Station
P.O. Box 128
San Clemente, CA 92674-0128

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
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 <u>Information Submitted in Support of the IFMP Review</u>

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

cc w/enclosure: Distribution via Listserv

DISTRIBUTION:
PUBLIC
RidsACRS_MailCTR Resource
MDusaniwskyj, NRR
RidsNrrDorlDpr Resource
RidsNrrLAPBlechman Resource
RidsRgn4MailCenter Resource
MPurdie, NRR
LPL4-2 r/f
EOlvera, NRR
RidsNrrDirsIfib Resource
RidsNrrDorlLpl4-2 Resource
RidsNrrPMSanOnofre Resource
MSampson, NMSS

ADAMS Accession No.: ML15182A256

OFFICE	NRR/DORL/LPL4-2/PM	NRR/DORL/LPL4-2/LA	NRR/DIRS/IFIB/BC	NMSS/DSFM/SFLB
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

OFFICIAL RECORD COPY

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

COMMISSIONERS:

Stephen G. Burns, Chairman
Kristine L. Svinicki
William C. Ostendorff
Jeff Baran

In the Matter of

DTE ELECTRIC COMPANY (Fermi Nuclear Power Plant, Unit 3)	**Docket No. 52-033-COL**
DTE ELECTRIC COMPANY (Fermi Nuclear Power Plant, Unit 2)	**Docket No. 50-341-LR**
DUKE ENERGY CAROLINAS, LLC (William States Lee III Nuclear Station, Units 1 and 2)	**Docket Nos. 52-018-COL 52-019-COL**
ENTERGY NUCLEAR OPERATIONS, INC. (Indian Point, Units 2 and 3)	**Docket Nos. 50-247-LR 50-286-LR**
FIRSTENERGY NUCLEAR OPERATING COMPANY (Davis-Besse Nuclear Power Station, Unit 1)	**Docket No. 50-346-LR**
FLORIDA POWER & LIGHT COMPANY (Turkey Point Nuclear Generating Plant, Units 6 and 7)	**Docket Nos. 52-040-COL 52-041-COL**

LUMINANT GENERATION COMPANY, LLC (Comanche Peak Nuclear Power Plant, Units 3 and 4)	**Docket Nos. 52-034-COL 52-035-COL**
NEXTERA ENERGY SEABROOK, LLC (Seabrook Station, Unit 1)	**Docket No. 50-443-LR**
NUCLEAR INNOVATION NORTH AMERICA LLC (South Texas Project, Units 3 and 4)	**Docket Nos. 52-012-COL 52-013-COL**
PACIFIC GAS AND ELECTRIC COMPANY (Diablo Canyon Nuclear Power Plant, Units 1 and 2)	**Docket Nos. 50-275-LR 50-323-LR**
PROGRESS ENERGY FLORIDA, INC. (Levy County Nuclear Power Plant, Units 1 and 2)	**Docket Nos. 52-029-COL 52-030-COL**
STP NUCLEAR OPERATING COMPANY (South Texas Project, Units 1 and 2)	**Docket Nos. 50-498-LR 50-499-LR**
TENNESSEE VALLEY AUTHORITY (Bellefonte Nuclear Power Plant, Units 3 and 4)	**Docket Nos. 52-014-COL 52-015-COL**
TENNESSEE VALLEY AUTHORITY (Sequoyah Nuclear Plant, Units 1 and 2)	**Docket Nos. 50-327-LR 50-328-LR**
TENNESSEE VALLEY AUTHORITY (Watts Bar Nuclear Plant, Unit 2)	**Docket No. 50-391-OL**
UNION ELECTRIC COMPANY (Callaway Plant, Unit 1)	**Docket No. 50-483-LR**

VIRGINIA ELECTRIC AND POWER COMPANY d/b/a DOMINION VIRGINIA POWER and OLD DOMINION ELECTRIC COOPERATIVE (North Anna Power Station, Unit 3)

Docket No. 52-017-COL

February 26, 2015

ATOMIC ENERGY ACT: CONTINUED STORAGE RULE; LICENSING; NUCLEAR REGULATORY COMMISSION AUTHORITY

The Commission is not required, under the Atomic Energy Act of 1954, as amended, to make predictive findings regarding the technical feasibility of spent fuel disposal as part of its reactor licensing decisions.

MEMORANDUM AND ORDER

Several environmental organizations in the captioned matters (collectively, Petitioners) have requested that we suspend final reactor licensing decisions pending our issuance of a "waste confidence safety decision."[1] Petitioners also have submitted companion filings proposing a new or amended waste confidence safety contention, together with related procedural motions to reopen the record in several of the captioned proceedings.[2] For the reasons set forth below, we deny

---

[1] *See, e.g.,* Petition to Suspend Final Decisions in All Pending Reactor Licensing Proceedings Pending Issuance of Waste Confidence Safety Findings (Sept. 29, 2014) (errata Oct. 1, 2014; amended and corrected petition Oct. 6, 2014 (Petition). Citations to the Petition in today's decision will reference the corrected Petition filed in the *Callaway* license renewal matter. A full list of the filings associated with this decision is set forth in the Appendix.

[2] *See, e.g.,* Missouri Coalition for the Environment's Motion for Leave to File a New Contention Concerning the Absence of Required Waste Confidence Safety Findings in the Relicensing Proceeding at Callaway 1 Nuclear Power Plant (Sept. 29, 2014) (Motion; filed in the Callaway license renewal docket). In some proceedings, petitioners also filed motions to reopen the record. *See, e.g.,* Motion to Reopen the Record for Callaway Nuclear Power Plant (Sept. 29, 2014) (Motion to Reopen; filed in the Callaway license renewal docket). Intervenors in the *Levy County* combined license proceeding filed a motion to reopen, but subsequently withdrew their motion. *See* Intervenors' Unopposed Motion to Withdraw Their Motion to Reopen the Record (Oct. 2, 2014); Order (Dismissing Environmental Waste Confidence Contention) (Oct. 1, 2014) (unpublished). With the withdrawal of this motion, nine motions to reopen remain pending before us. In the *Indian Point* license renewal proceeding, *(Continued)*

the suspension petitions, decline to admit the related contention, and deny the motions to reopen.

Petitioners primarily assert that the Atomic Energy Act of 1954, as amended (the Act), requires the NRC, as a precondition to issuing or renewing operating licenses for nuclear power plants, to make definitive findings concerning the technical feasibility of a repository for the disposal of spent nuclear fuel. We rejected a nearly identical argument in 1977 and, though much of the regulatory framework has changed in the intervening years, our reading of the Act has not.[3]

Our conclusion that a suspension is not warranted finds support not only in our interpretation of the Act itself, but also in the regulatory authority that Congress has provided to the agency to protect public health and safety. Indeed, our confidence in the safety and technical feasibility of systems for the storage and disposal of spent fuel has only increased since the late 1970s, as demonstrated by our expanded regulatory scheme and the ongoing licensing of such systems, as well as the efforts that are under way — both in the United States and abroad — to develop repositories for the disposal of spent fuel. Thus, today we not only address Petitioners' concerns, but we also take the opportunity to confirm the continued validity of our determinations regarding the technical feasibility of safe spent fuel storage and ultimate disposal in a repository.

I. BACKGROUND

Recently, we approved a final rule and generic environmental impact statement, issued in accordance with the National Environmental Policy Act (NEPA) and the Administrative Procedure Act, to address the environmental impacts associated with the storage of spent nuclear fuel after the end of a reactor's license term (the Continued Storage Rule).[4] Following the publication of the Continued Storage Rule and supporting generic environmental impact statement (Continued Storage

---

Riverkeeper filed a substantively identical suspension petition together with a motion transmitting a new contention a few days after the initial suspension petitions were filed. Petition to Suspend Final Decision in Indian Point Relicensing Proceeding Pending Issuance of Waste Confidence Safety Findings (Oct. 3, 2014); Riverkeeper Consolidated Motion for Leave to File a New Contention and New Contention RK-10 Concerning the Absence of Required Waste Confidence Safety Findings (Oct. 3, 2014).

[3] *See* Natural Resources Defense Council, Denial of Petition for Rulemaking, 42 Fed. Reg. 34,391, 34,393 (July 5, 1977), *aff'd, Natural Resources Defense Council, Inc. v. NRC,* 582 F.2d 166 (2d Cir. 1978) (NRDC PRM Denial).

[4] Final Rule: "Continued Storage of Spent Nuclear Fuel," 79 Fed. Reg. 56,238 (Sept. 19, 2014) (Continued Storage Rule); NUREG-2157, "Generic Environmental Impact Statement for Continued Storage of Spent Nuclear Fuel," Vols. 1 & 2 (Sept. 2014) (ADAMS Accession Nos. ML14196A105 and ML14196A107) (Continued Storage GEIS).

GEIS), Petitioners filed substantively identical petitions to suspend final licensing decisions, related motions requesting the admission of new — or, in one instance, amended — contentions in the captioned matters, and, in several proceedings, motions to reopen the proceedings to consider the proposed contentions.[5]

Exercising our inherent supervisory authority over agency proceedings, we took review of the petitions and motions ourselves and set a briefing schedule.[6] All answers oppose the suspension petitions and admission of the accompanying contention.[7] Petitioners filed a consolidated reply.[8]

Petitioners claim that we cannot satisfy our statutory responsibilities under the Atomic Energy Act and that we no longer have a lawful basis for issuing initial and renewed licenses for nuclear power reactors.[9] They assert that we must, therefore, suspend final licensing decisions unless and until we make a "safety finding" associated with disposal.[10] Petitioners ask us to admit the following contention:

> The NRC lacks a lawful basis under the Atomic Energy Act . . . for issuing or renewing an operating license in this proceeding because it has not made currently valid findings of confidence or reasonable assurance that the hundreds of tons of highly radioactive spent fuel that will be generated during any reactor's 40-year license term or 20-year license renewal term can be safely disposed of in a repository. The NRC must make these predictive safety findings in every reactor

---

[5] *See, e.g.,* Petition, and Motion to Reopen.

[6] CLI-14-9, 80 NRC 147 (2014).

[7] *See, e.g.,* NRC Staff Consolidated Answer to Petitions to Suspend Final Reactor Licensing Decisions, Motions to Admit a New Contention, and Motions to Reopen the Record (Oct. 31, 2014); Entergy's Combined Answer to Riverkeeper's Proposed New Contention RK-10 and Petition to Suspend Final License Renewal Decision Pending Issuance of Waste Confidence "Safety" Findings (Oct. 31, 2014); Tennessee Valley Authority's Answer Opposing Petition to Suspend Final Decisions in All Pending Reactor Licensing Proceedings Pending Issuance of Waste Confidence Safety Findings and Motions for Leave to File New Contention (Oct. 31, 2014); Tennessee Valley Authority's Answer to Motion to Reopen the Record for Sequoyah Nuclear Power Plant and Motion to Reopen the Record for Bellefonte Nuclear Power Plant (Oct. 31, 2014) (TVA Answer to Motions to Reopen).

[8] Petitioners' and Intervenors' Consolidated Reply to Answers to Petitions to Suspend Final Reactor Licensing Decisions, Motions to Admit a New Contention, and Motions to Reopen the Record (Nov. 7, 2014) (Reply). In addition, the Nuclear Energy Institute filed an unopposed motion for leave to file a brief *amicus curiae* opposing the Petition. Nuclear Energy Institute, Inc.'s Motion for Leave to File Amicus Curiae Brief (Oct. 31, 2014); Amicus Curiae Brief of the Nuclear Energy Institute, Inc. in Response to Suspension Petitions and Waste Confidence Safety Contentions (Oct. 31, 2014). Our rule governing *amicus curiae* participation does not contemplate a brief under the current circumstances. *See* 10 C.F.R. § 2.315(d) (providing for *amicus* filings at our discretion under 10 C.F.R. § 2.341 or *sua sponte*). We, nonetheless, have considered the Nuclear Energy Institute's views as a matter of discretion. *See, e.g., Southern California Edison Co.* (San Onofre Nuclear Generating Station, Units 2 and 3), CLI-13-9, 78 NRC 551, 556 n.17 (2013).

[9] *See, e.g.,* Motion at 3.

[10] *See, e.g.,* Petition at 8 (unnumbered).

> licensing decision in order to fulfill its statutory obligation under the [Act] to protect public health and safety from the risks posed by irradiated reactor fuel generated during the reactor's license term.[11]

Petitioners' contention, which comes on the heels of our issuance of the Continued Storage Rule, relies in large part on the fact that, unlike prior versions of the Rule, the Continued Storage Rule is no longer supported by specific "findings" concerning, among other things, reasonable assurance of the feasibility of a repository. To provide a more complete understanding of the context of Petitioners' argument, we provide a brief history of our "waste confidence" proceedings.[12]

In 1976, the Natural Resources Defense Council (NRDC) filed a petition requesting that we conduct a rulemaking to determine whether spent fuel "can be generated in nuclear power reactors and subsequently disposed of without undue risk to the public health and safety."[13] NRDC argued that, without this determination, we should refrain from making final decisions on "pending or future requests for operating licenses."[14] We denied NRDC's petition and found that, as a matter of statutory interpretation, the Atomic Energy Act did not require us to make the requested finding.[15] In the denial, we noted the NRC's obligations with respect to spent fuel storage and disposal at the time of a reactor licensing decision. Specifically, we explained that, at the time a license is issued, we must "be assured that the wastes generated by licensed power reactors can be safely handled and stored as they are generated."[16] As part of the reactor licensing process, we noted, an applicant must submit information to allow the NRC to "assure that the design provides for safe methods for interim storage of spent nuclear fuel."[17] Given the focus during the licensing process on the safety of licensed operations, we determined that the text of the Atomic Energy Act (combined with Congress's understanding of the state of the development of a repository) did not require us to make, as a precondition to licensing, an express

---

[11] Motion at 3-4 (citations omitted).

[12] A complete history of the prior waste confidence proceedings can be found in Chapter 1 of the Continued Storage GEIS.

[13] NRDC PRM Denial, 42 Fed. Reg. at 34,391.

[14] *Id.*

[15] *Id.*

[16] *Id.* Today, this assurance is demonstrated by compliance with our regulations that govern the safe storage of spent fuel. *See, e.g.,* Domestic Licensing of Production and Utilization Facilities, 10 C.F.R. Part 50 (2014) and General License for Storage of Spent Fuel at Power Reactor Sites, 10 C.F.R. Part 72, Subpart K (2014), which grants a general license to all Part 50 and Part 52 reactor licensees to store spent fuel in an independent spent fuel storage installation.

[17] NRDC PRM Denial, 42 Fed. Reg. at 34,391.

determination that spent fuel generated during operation could be disposed of safely.[18]

The denial also included a separate statement of policy.[19] In that discussion, which Petitioners reference throughout their filings, we stated that we would not continue to license reactors if we "did not have reasonable confidence that . . . [spent fuel] can and will in due course be disposed of safely."[20] We explained that our "implicit" finding that methods of safe permanent storage were available could be "readily distinguished" from the type of safety findings that the agency is called upon to make during the course of reactor licensing under the Atomic Energy Act and that any finding in this regard "would not have to be a definitive conclusion that permanent disposal of high-level wastes can be accomplished safely at the present time."[21]

NRDC sought judicial review of the petition denial. The Court of Appeals for the Second Circuit affirmed the denial and endorsed our conclusion that the Atomic Energy Act does not, as a prerequisite to licensing, require a finding of reasonable assurance that "highly hazardous and long-lived radioactive materials can be disposed of safely."[22] The court concluded that, by seeking to require an express finding concerning safe disposal prior to licensing, "NRDC simply reads too much into the [Atomic Energy Act] . . . . We are satisfied that Congress did not intend such a condition."[23]

In addition to recognizing that the text of the Atomic Energy Act does not mandate such a specific finding, the court relied on Congress's decades-long tacit approval of nuclear power plant licensing even in the absence of a disposal site.[24] Further, the court explained, if NRDC's view of the Atomic Energy Act were correct, it would be "incredible that AEC and its successor NRC would have been violating the [Act] for almost twenty years with no criticism or statutory amendment by Congress, which has been kept well informed of [disposal] developments."[25] Accordingly, the court quoted favorably that it was "fair to read this history as a *[d]e facto* acquiescence in and ratification of the Commission's licensing procedure by Congress."[26]

---

[18] *Id.* at 34,391-93.

[19] *Id.* at 34,393-94.

[20] *Id.* at 34,393.

[21] *Id.*

[22] *NRDC,* 582 F.2d at 171.

[23] *Id.*

[24] *Id.* at 173-74. The court found Congress's silence in the face of ongoing reactor licensing "deafening." *Id.* at 171.

[25] *Id.*

[26] *Id.* at 172 (quoting *Power Reactor Development Co. v. International Union of Electrical, Radio & Machine Workers,* 367 U.S. 396, 409 (1961)).

The court did not rest its decision solely on the legislative history of the Act or on tacit congressional approval of reactor licensing absent safety findings for a repository. "[I]f there were any doubt over the intent of Congress" not to require a safety finding on spent fuel disposal, explained the court, it was "persuaded that the matter was laid to rest by enactment of the Energy Reorganization Act of 1974."[27] The court noted that, in that act, "Congress expressly recognized and impliedly approved NRC's regulatory scheme and practice under which the safety of interim storage of [spent fuel] at commercial nuclear power reactor sites has been determined separately from the safety of . . . permanent storage facilities which have not, as yet, been established."[28] Since the passage of the Energy Reorganization Act of 1974 as well as the Second Circuit's decision in *NRDC v. NRC,* Congress has had numerous opportunities to consider our interpretation of the Atomic Energy Act with respect to a disposal safety finding at the time of reactor licensing. But in each case, Congress has left intact both this agency's and the court's interpretation.[29]

Since 1984, we have completed four rulemaking proceedings that analyzed the environmental impacts of the continued storage of spent fuel after the end of a reactor's license term (the "waste confidence" and "continued storage" proceedings).[30] The first rulemaking, the 1984 waste confidence proceeding, was prompted by a remand from the Court of Appeals for the District of Columbia Circuit in *Minnesota v. NRC.*[31] In that case, the petitioners challenged the NRC's approval of amendments to the Prairie Island and Vermont Yankee nuclear power plant operating licenses to allow for the use of higher-density spent-fuel-storage racks in the reactors' spent fuel pools.[32] The court observed that the Second Circuit

---

[27] *Id.* at 174 (citations omitted).

[28] *Id.* The court observed that, in considering passage of the 1974 legislation, Congress heard testimony from scientists and other representatives of groups "urg[ing] Congress, unsuccessfully, to halt further commercial power plant licensing pending resolution of the waste disposal issue." *Id.* at 171 n.9, 174-75 (citations omitted).

[29] *See, e.g.,* Nuclear Waste Policy Act of 1982, Pub. L. No. 97-425, 96 Stat. 2201 (1982); Energy Policy Act of 2005, Pub. L. 109-58, 119 Stat. 594 (2005).

[30] Final Waste Confidence Decision, 49 Fed. Reg. 34,658 (Aug. 31, 1984 (1984 Waste Confidence Decision); Requirements for Licensee Actions Regarding the Disposition of Spent Fuel upon Expiration of Reactor Operating Licenses, 49 Fed. Reg. 34,688 (Aug. 31, 1984) (1984 Temporary Storage Rule); Consideration of Environmental Impacts of Temporary Storage of Spent Fuel After Cessation of Reactor Operation, 55 Fed. Reg. 38,472 (Sept. 18, 1990) (1990 Temporary Storage Rule); Waste Confidence Decision Review, 55 Fed. Reg. 38,474 (Sept. 18, 1990) (1990 Waste Confidence Decision); Consideration of Environmental Impacts of Temporary Storage of Spent Fuel After Cessation of Reactor Operation, 75 Fed. Reg. 81,032 (Dec. 23, 2010) (2010 Temporary Storage Rule); Waste Confidence Decision Update, 75 Fed. Reg. 81,037 (Dec. 23, 2010) (2010 Waste Confidence Decision); Continued Storage GEIS; and Continued Storage Rule.

[31] *Minnesota v. NRC,* 602 F.2d 412 (D.C. Cir. 1979).

[32] *Id.* at 412.

had recently ruled in *NRDC v. NRC* that "Congress did not intend in enacting the Atomic Energy Act to require a demonstration that nuclear wastes could safely be disposed of before licensing of nuclear plants was permitted," and it did not disagree with that result.[33] Referring to the language in the policy statement accompanying the denial of the petition for rulemaking, the court directed the NRC to determine "whether there is reasonable assurance that an off-site storage solution will be available by [the end of a reactor's license term], and if not, whether there is reasonable assurance that the fuel can be stored safely at the sites beyond those dates."[34]

In 1984, we published our first Waste Confidence Decision and Temporary Storage Rule. The Waste Confidence Decision included "findings," expressed in terms of "reasonable assurance," that, among other things, a repository was technically feasible, one could be open by 2007-2009, and the spent fuel could be safely stored for 30 years after the end of a reactor's license term.[35] In 1990, we revisited the Decision and Temporary Storage Rule and updated the findings to reflect a new expected date for a repository to become available ("the first quarter of the twenty-first century") and to include a 30-year license renewal term in our safe-storage analysis.[36] In 2010, we issued another update that removed the anticipated date for repository availability (explaining instead that a repository would be available "when necessary") and expanded the safe-storage analysis time frame from 30 years after the end of the reactor's license term to 60 years after the end of the reactor's license term.[37]

Several states, an Indian Tribe, and environmental organizations (some of whom are Petitioners here) filed suit before the Court of Appeals for the District of Columbia Circuit challenging the 2010 update to the Decision and Temporary Storage Rule. In 2012, in *New York v. NRC,* the court vacated and remanded the decision and rule, and found that we had not satisfied our obligations under NEPA with respect to three issues: (1) we did not consider the environmental impacts of a repository never becoming available; (2) our analysis of spent fuel

---

[33] *Id.* at 417 (citing *NRDC,* 582 F.2d at 166).

[34] *Id.* at 418. In reaching this decision, the court recognized the long-term nature of the concerns associated with spent fuel storage and disposal when it declined to vacate the license amendments that were the subject of the case, noting that doing so "would effectively shut down the plants." *Id.* Moreover, its decision was predicated on the context of the particular license amendments at issue — to allow high-density spent fuel storage; in fact, the court acknowledged the Second Circuit's ruling in *NRDC v. NRC* and did not disagree with that result. *See id.* at 417.

[35] 1984 Waste Confidence Decision, 49 Fed. Reg. at 34,659-60; 1984 Temporary Storage Rule, 49 Fed. Reg. at 34,688.

[36] *See, e.g.,* 1990 Temporary Storage Rule, 55 Fed. Reg. at 38,473; 1990 Waste Confidence Decision, 55 Fed. Reg. at 38,503-04.

[37] *See, e.g.,* 2010 Temporary Storage Rule, 75 Fed. Reg. at 81,037; 2010 Waste Confidence Decision, 75 Fed. Reg. at 81,038.

pool leaks was not forward-looking; and (3) we had not sufficiently considered the consequences of spent fuel pool fires.[38] The court did not specifically address any issues arising under the Atomic Energy Act.

Following the court's decision in *New York,* we suspended all final decisions for licenses that relied on the Waste Confidence Decision and Temporary Storage Rule.[39] Shortly thereafter we directed the NRC Staff to prepare a generic environmental impact statement to support an updated rule and address the deficiencies that the court identified.[40] We approved the final Continued Storage GEIS and Rule, now known as the Continued Storage Rule, in September 2014.[41] Although it did not include the discrete findings made in the waste confidence proceedings, and although it did not express our conclusions in terms of "reasonable assurance," the Continued Storage GEIS contains a comprehensive discussion supporting our unqualified conclusion that both safe storage and disposal in a repository are technically feasible.[42]

Thus, while much has changed since we last addressed the specific issue raised in Petitioners' contention, much has stayed the same. In each of our waste confidence proceedings, as well as in the recently concluded continued storage proceeding, we determined that deep geologic disposal of spent nuclear fuel is technically feasible.[43] Similarly, throughout our rulemakings conducted over the past 30 years, neither we nor the courts have questioned our initial conclusion that the Atomic Energy Act does not require the explicit "reasonable assurance" finding requested by Petitioners. And of course, our licensing has proceeded on the basis of these well-settled premises.

II. DISCUSSION

With this background in mind, we turn to the petitions at hand. Petitioners claim a deficiency in our ability to satisfy our basic licensing responsibilities under the Atomic Energy Act, which Petitioners believe results in the loss of

---

[38] *New York v. NRC,* 681 F.3d 471, 473, 481-82 (D.C. Cir. 2012).

[39] *Calvert Cliffs 3 Nuclear Project, LLC* (Calvert Cliffs Nuclear Power Plant, Unit 3), CLI-12-16, 76 NRC 63, 66-67 (2011).

[40] Staff Requirements — COMSECY-12-0016 — Approach for Addressing Policy Issues Resulting from Court Decision to Vacate Waste Confidence Decision and Rule (Sept. 6, 2012) (ADAMS Accession No. ML12250A032).

[41] Staff Requirements — Affirmation Session 10:00 a.m., Tuesday, August 26, 2014, Commissioners' Conference Room, One White Flint North, Rockville, Maryland (Open to Public Attendance) (Aug. 26, 2014) (ADAMS Accession No. ML14237A092).

[42] *See generally* Continued Storage GEIS, App. B.

[43] *Compare* 1984 Waste Confidence Decision, 49 Fed. Reg. at 34,659, *with* 1990 Temporary Storage Rule, 55 Fed. Reg. at 38,472, *and with* Continued Storage GEIS § B.2.1.

our "lawful basis for licensing or relicensing nuclear reactors."[44] This claim is distinguishable from those raised in the suspension petitions that we have considered in recent years. Following the events of September 11, 2001, and again following the accident at Fukushima Dai-ichi, petitioners asserted that our actions were insufficient to satisfy our general obligation under the Atomic Energy Act to protect public health and safety.[45] Here, on the other hand, Petitioners claim that we have an obligation under the Atomic Energy Act to make explicit findings regarding the safety of spent fuel disposal as a prerequisite to our reactor licensing decisions.[46] As such, our usual framework for considering suspension requests is not applicable to the case at hand. Instead, exercising our inherent supervisory authority over agency proceedings, we consider Petitioners' claims regarding the scope of our obligations under the Atomic Energy Act. As discussed below, we find Petitioners' Atomic Energy Act claims to be without merit, and we therefore deny the petitions and the companion proposed contention and motions to reopen.[47]

Together with the Energy Reorganization Act of 1974, the Atomic Energy Act provides the basis for our authority to regulate the use of special nuclear material in facilities like nuclear power reactors.[48] We can issue nuclear power reactor licenses to applicants only upon a finding that "the utilization . . . of special nuclear material will be in accord with the common defense and security and will provide adequate protection to the health and safety of the public."[49] An applicant

[44] Reply at 11.

[45] *See, e.g., Private Fuel Storage, L.L.C.* (Independent Spent Fuel Storage Installation), CLI-01-26, 54 NRC 376, 380 (2001); *Union Electric Co.* (Callaway Plant, Unit 2), CLI-11-5, 74 NRC 141, 151 (2011).

[46] Reply at 11. As Petitioners acknowledge, "the Petition is *not* a motion for a stay of the effectiveness of a decision pursuant to 10 C.F.R. § 2.342 or any other kind of request for equitable relief." *Id.* (emphasis in original). *See generally* 10 C.F.R. § 2.342 (governing stays of the actions or decisions of a presiding officer pending filing of a petition for review).

[47] Because Petitioners' Atomic Energy Act claim fails, they have not raised an issue material to findings that the NRC must make to support final decisions in the captioned matters and they are unable to satisfy our contention admissibility standards or meet the criteria to reopen a closed record. *See* 10 C.F.R. §§ 2.309(f)(1) and 2.326. We therefore decline to admit Petitioners' proposed contention and deny their motions to reopen. Moreover, we deny as moot Blue Ridge Environmental Defense League's motions to reopen in the *Sequoyah* and *Bellefonte* proceedings because those proceedings remain open. *See* TVA Answer to Motion to Reopen at 1.

[48] Atomic Energy Act of 1954, 42 U.S.C. §§ 2011-2297h-13 (2012) and Energy Reorganization Act of 1974, 42 U.S.C. §§ 5801-5891 (2012).

[49] Atomic Energy Act § 182a, 42 U.S.C. § 2232 (2012).

As we noted in the Continued Storage GEIS, Congress "authorized and directed the NRC to issue regulations establishing requirements for providing adequate protection to public health and safety and common defense and security (see Atomic Energy Act [§] 161b) . . . . [U]nder current law, the *(Continued)*

demonstrates its ability to meet these standards, and thus its entitlement to a license, by submitting a license application that satisfies our licensing criteria.[50] If a power reactor license applicant is unable to meet our regulatory requirements or if we find that the proposed use of special nuclear material will not be in accord with the common defense and security or will not provide adequate protection of public health and safety, then we will not issue a license.[51]

Petitioners argue that part of this analysis must include a "safety" or "waste confidence" finding regarding the technical feasibility of a deep geologic repository for the disposal of spent fuel generated at nuclear power plants.[52] Petitioners contend that without such a finding we are unable to make the required finding of adequate protection under the Atomic Energy Act and must, therefore, refrain from issuing licenses until this finding is made.[53] Further, Petitioners argue, this safety finding must be supported by a separate NEPA analysis of the environmental impacts of spent fuel disposal — either in the form of an environmental impact statement or an environmental assessment.[54]

A. Petitioners' Atomic Energy Act Claims

Petitioners argue that the NRC's historic practice, the plain language of the Atomic Energy Act, and relevant case law support their claims. We disagree. At no time have we, Congress, or the courts articulated the view that the Atomic Energy Act requires a "finding" or "predictive safety findings" regarding the disposal of spent fuel in a repository as a prerequisite to issuing a nuclear reactor license. We see no reason to alter our long-standing interpretation of the Atomic Energy Act.

Our interpretation of the agency's obligations under the Atomic Energy Act with respect to spent fuel disposal began with our 1977 denial of NRDC's petition for rulemaking.[55] We found then that the Atomic Energy Act does not require us

---

NRC will issue a nuclear power plant or materials license (including a license authorizing storage of spent fuel) when the NRC determines that a license applicant has met the NRC's regulatory standards for issuance of a license, addressing adequate protection of public health and safety and common defense and security, and the NRC has no reason to doubt that issuance of the license would provide adequate protection." Continued Storage GEIS § 1.6.2.1.

[50] *See, e.g.,* 10 C.F.R. Parts 50, 52, and 54.

[51] *See, e.g., Maine Yankee Atomic Power Co.* (Maine Yankee Atomic Power Station), ALAB-161, 6 AEC 1003, 1007 (1973) ("Unless the safety findings prescribed by the Atomic Energy Act and the regulations can be made, the reactor does not obtain a license — no matter how badly it is needed.").

[52] Motion at 3-4.

[53] Petition at 2-3 (unnumbered).

[54] Motion to Reopen at 4. Among other things, Petitioners argue that this NEPA analysis must consider the costs of spent fuel storage and disposal. *Id.*

[55] NRDC PRM Denial, 42 Fed. Reg. at 34,391-92.

to make a finding regarding spent fuel disposal as part of our reactor licensing decisions.[56] And the Second Circuit endorsed our construction of the Act:

> [W]e hold that NRC is not required to conduct the rulemaking proceeding requested by NRDC or to withhold action on pending or future applications for nuclear power reactor operating licenses until it makes a determination that high-level radioactive wastes can be permanently disposed of safely.[57]

Both our denial of the petition for rulemaking and the court's affirmance of this decision were grounded in the language of Atomic Energy Act sections 103, 161, and 182 — the very sections relied upon here by Petitioners. As the court expressly concluded in *NRDC,* we find that Petitioners read "too much into the [Act]."[58]

Section 103d prohibits the agency from issuing a license if doing so "would be inimical to the common defense and security or the health and safety of the public."[59] Petitioners claim that the "plain language" of this section conflicts with the interpretation of the Atomic Energy Act that we adopted in the denial of NRDC's petition for rulemaking. Specifically, they take issue with our conclusion that "the statutory findings required by section 103 apply specifically to the 'proposed activities' and 'activities under such licenses'" but do not apply to disposal activities that might result from the operation of a licensed facility.[60] Section 103 does not contemplate consideration of spent fuel disposal in the NRC's licensing decisions, and we decline to infer from Congress's silence an affirmative obligation to the contrary.[61]

The same is true of the other Atomic Energy Act provisions upon which Petitioners rely. Section 161 establishes the general scope of the NRC's authority, yet nowhere does it discuss spent fuel disposal.[62] Similarly, section 182 specifies

---

[56] *Id.*

[57] *NRDC,* 582 F.2d at 175.

[58] *Id.* at 171.

[59] Atomic Energy Act, § 103, 42 U.S.C. § 2133 (2012).

[60] Motion at 6-7; NRDC PRM Denial, 42 Fed. Reg. at 34,391.

[61] *See NRDC,* 582 F.2d at 170-71. Petitioners also rely on the concurring opinion of Judge Tamm from *Minnesota v. NRC.* In his concurrence, Judge Tamm noted his "belief that section 102(2)(C) of [NEPA] and section 103(d) [of the Act] . . . mandate the determination that the Commission identified in" the NRDC PRM Denial. *Minnesota,* 602 F.2d at 419 (Tamm, J., concurring). But the majority did not express this view, and a concurring opinion, by its nature, does not carry the force of law, except in very narrow circumstances not applicable here. *See generally United States v. Duvall,* 740 F.3d 604, 605 (D.C. Cir. 2013). Had a majority of the Court in *Minnesota* agreed with Judge Tamm's expansive view of our Atomic Energy Act obligations, these views would have been reflected in the majority opinion.

[62] Atomic Energy Act, § 161, 42 U.S.C. § 2201 (2012).

the information that must be provided by an applicant for a license with no reference to spent fuel disposal.[63] Thus, the text of the Atomic Energy Act does not compel the conclusion that we are required to include "findings" regarding spent fuel disposal in our reactor licensing decisions, and we decline to interpret it otherwise. And, in light of our interpretation, the related NRC regulations do not require information about the eventual disposal of the spent fuel that would be generated by the reactor.[64]

Moreover, as the Second Circuit explained in *NRDC,* the conclusion that the Atomic Energy Act does not require "safety findings" is further supported by the legislative history of the Act and subsequent congressional action. For example, in 1959, Congress held hearings regarding the disposal of spent nuclear fuel and, at that time, Congress "was made aware of the fact that the problem of permanent disposal of high-level waste had not been solved."[65] But Congress did not restrict or modify the NRC's licensing authority. Further, Congress later approved a continuation of the licensing approach in the Atomic Energy Act when it transferred the licensing functions of the Atomic Energy Commission to us via the Energy Reorganization Act of 1974.[66] Had Congress believed that our licensing activities required the finding sought by Petitioners, it could have enacted legislation consistent with this understanding at any time between 1954 and today.[67] That Congress has maintained this course despite our rejection of NRDC's interpretation of the Atomic Energy Act in the denial of the petition for rulemaking, the Second Circuit's endorsement of our construction of the Act in *NRDC,* and the numerous opportunities for legislative clarification provides further confirmation of the propriety of our interpretation of the Act.[68]

Petitioners rely heavily upon our statement, expressed as part of the policy discussion included in the denial of NRDC's petition for rulemaking, that we would not continue to license reactors if we "did not have reasonable confidence that . . . [spent fuel] can and will in due course be disposed of safely."[69] They assert that this statement should guide our interpretation of the Act and that any

---

[63] Atomic Energy Act, § 182, 42 U.S.C. § 2232 (2012).

[64] *See, e.g., id.;* 10 C.F.R. Parts 50, 52, and 54 (2014).

[65] NRDC PRM Denial, 42 Fed. Reg. at 34,392 (citing "Industrial Radioactive Waste Disposal," Hearings Before the JCAE Special Subcommittee on Radiation, Jan. 29-30, Feb. 2-3, and July 29, 1959, 86th Cong., 1st Sess. (1959)).

[66] Energy Reorganization Act of 1974, Pub. L. 93-438, 88 Stat. 1233 (1974).

[67] *See, e.g.,* Nuclear Waste Policy Act of 1982, Pub. L. No. 97-425, 96 Stat. 2201 (1982); Energy Policy Act of 2005, Pub. L. 109-58, 119 Stat. 594 (2005).

[68] Indeed, in recent years, numerous congressional hearings over the funding of the Yucca Mountain repository have highlighted the absence of a national consensus on siting a repository.

[69] NRDC PRM Denial, 42 Fed. Reg. at 34,393.

acquiescence by Congress in our interpretation was conditioned on its existence.[70] But in the NRDC PRM Denial we expressly distinguished findings of the kind contemplated by the Atomic Energy Act and the NRC's licensing regulations from the more generalized conclusion in the policy statement.[71] As we explained at the time:

> Even if, contrary to the Commission's view, some kind of prior finding on waste disposal safety were required under the statutory scheme, such a finding would not have to be a definitive conclusion that permanent disposal of high-level wastes can be accomplished safely at the present time. There is no question that prior to authorizing operation of a reactor the Commission must find pursuant to section 182 that hazards which become fully mature with start-up will be dealt with safely from the beginning. *But the quality of this reactor safety finding can be readily distinguished from the quality of findings regarding impacts on public health and safety which will not mature until much later, if ever.* The hazards associated with permanent disposal will become acute only at some relatively distant time when it might be no longer feasible to store radioactive wastes in facilities subject to surveillance.[72]

It was only after this discussion that we added: "The Commission would not continue to license reactors if it did not have reasonable confidence that the wastes can and will in due course be disposed of safely."[73] Moreover, we pointed out that the program for siting and developing a geologic repository was not within the NRC's statutory responsibilities under the Atomic Energy Act, another reason rendering an explicit safety finding on spent fuel disposal inappropriate.[74]

When considered within the context of our denial of the petition for rulemaking, it is clear that the statement at issue was nothing more than what it purported to be: a statement of our policy regarding the licensing of nuclear power plants and our confidence in the availability of a disposal solution.[75] This policy has always existed independent of our legal conclusion that no obligation exists under the Atomic Energy Act to make predictive findings regarding spent fuel disposal as part of our reactor licensing decisions.

---

[70] *See, e.g.,* Reply at 7.

[71] NRDC PRM Denial, 42 Fed. Reg. at 34,393.

[72] *Id.* (emphasis added).

[73] *Id.*

[74] In this regard, we observed that the Energy Research and Development Administration (the Department of Energy's predecessor agency) was responsible for the development of a high-level waste repository; the NRC's statutory responsibilities "to insure that permanent disposal of high-level radioactive wastes will be accomplished safely" were, and still are, limited to licensing the repository. *Id.*

[75] *Id.*

Petitioners also misapprehend the relevant case law. Specifically, Petitioners misread the Second Circuit's opinion in *NRDC v. NRC,* the only court decision to have directly addressed the issue. Overlooking the express holding that endorsed our interpretation of the Act,[76] Petitioners instead quote the court's characterization of our policy and practice: "[The] NRC maintains that . . . its long-continued regulatory practice of issuing operating licenses, with an implied finding of reasonable assurance that safe permanent disposal of [spent nuclear fuel] can be available when needed, is in accord with the intent of Congress underlying the [Atomic Energy Act] and [Energy Reorganization Act]."[77] But that description neither constitutes the court's holding nor reflects an admission concerning our interpretation of our statutory obligations. Rather, it reflects our view that our practice was consistent with the conclusion that a specific finding of repository feasibility was not a prerequisite under the Atomic Energy Act to reactor licensing. And the court agreed: "Congress expressly recognized and impliedly approved NRC's regulatory scheme and practice under which the safety of interim storage of high-level wastes at commercial nuclear power reactor sites has been determined separately from the safety of Government-owned permanent storage [disposal] facilities which have not, as yet, been established."[78]

Petitioners also rely on two subsequent decisions by the D.C. Circuit, *New York v. NRC* and *Minnesota v. NRC.* But in neither of these cases did the court find a statutory obligation on the part of the NRC to prepare "waste confidence" safety findings prior to or as part of our reactor licensing decisions. In *New York,* the court did not consider Atomic Energy Act issues. Instead, the remand was based solely on the court's finding that we did not satisfy our obligations under NEPA.[79]

In *Minnesota,* the court remanded for our consideration the question "whether there is reasonable assurance that an off-site storage solution will be available by . . . the expiration of the plants' operating licenses, and if not, whether there is reasonable assurance that the [spent] fuel can be stored safely at the sites beyond those dates."[80] Further, as distinct from the concurrence, the court majority refrained from identifying an obligation to make findings under the Atomic Energy Act. In that regard, the court expressly declined to "set aside or

---

[76] *NRDC,* 582 F.2d at 175 ("[W]e hold that NRC is not required to conduct the rulemaking proceeding requested by NRDC or to withhold action on pending or future applications for nuclear power reactor operating licenses until it makes a determination that high-level radioactive wastes can be permanently disposed of safely.").

[77] *Id.* at 170.

[78] *Id.* at 174.

[79] *New York,* 681 F.3d at 471, 483.

[80] *Minnesota,* 602 F.2d at 418.

stay the challenged license amendments,"[81] thus confirming that the court did not view the amendments to be contingent upon any additional safety determination under the Atomic Energy Act.

To be sure, our "findings" in the initial waste confidence proceeding likely caused some confusion. We understand that because of how they were framed, they could have been, and likely were, interpreted by some as safety findings made under and compelled by the Atomic Energy Act. That we responded to the *Minnesota* remand as we did, however, does not mean that the particular form of our response was compelled by the Atomic Energy Act. Rather, the formal "findings" in the initial waste confidence proceeding resulted from our use of a hybrid rulemaking proceeding, which combined elements of a formal "on the record" proceeding with the more common "notice and comment" rulemaking widely used today.[82] Formal rulemakings often result in "findings," such as the ones we made in our first waste confidence proceeding.[83] Moreover, that approach made sense at the time, which was long before our framework for regulating the safe storage and disposal of spent fuel had matured into its current state, and long before we had comprehensively evaluated the environmental impacts of the storage of spent nuclear fuel for an extended time frame — a task we now have completed in the Continued Storage GEIS.

Throughout their motions, Petitioners ascribe significance to our failure to use the term "reasonable assurance" to describe the extent of our consideration of the technical feasibility of disposal.[84] But as the technical agency entrusted by Congress to make determinations of this sort, we have concluded — without qualification — that a geologic repository is technically feasible.[85] As we acknowledged in the Continued Storage GEIS, the uncertainty in spent fuel disposal lies not with the technical feasibility of long-term storage and disposal, but with the political and societal factors that continue to delay the construction of a repository.[86] We recognized this uncertainty in the Continued Storage GEIS by analyzing the possibility that a repository will never become available.[87] Our decision today is consistent with our long-standing conclusion.

Finally, it bears repeating that our recently completed Continued Storage GEIS considers the issues raised by Petitioners. Many of the groups petitioning us now provided essentially identical comments as part of our recently completed

---

[81] *Id.* at 413.
[82] *See* 1984 Waste Confidence Decision, 49 Fed. Reg. at 34,658-60.
[83] *See id.*
[84] *See, e.g.,* Reply at 9-10.
[85] Continued Storage GEIS § B.2.1.
[86] *Id.*
[87] *See, e.g., id.* § 1.8.2.

Continued Storage proceeding.[88] We responded to Petitioners' comments in the final GEIS and nothing has changed since then that would cause us to question the technical feasibility of disposal in a repository — safe geologic disposal is achievable with currently available technology.[89] Our analysis in the Continued Storage GEIS builds on decades of experience and multiple rulemaking proceedings.[90] Specifically, our conclusion finds support in ongoing research in the United States and abroad, along with the ability to characterize and quantitatively assess the capabilities of geologic and engineered barriers, experience gained from the Staff's review of the Department of Energy's construction authorization application for a repository at Yucca Mountain, disposal activities at the Waste Isolation Pilot Plant, and continued progress toward a repository in other countries.[91] Indeed, contrary to the situation that accompanied the issuance of the initial Waste Confidence Decision, our regulatory framework now includes specific standards and requirements for licensing the storage of spent fuel and, in the case of Yucca Mountain, standards for licensing a repository.[92]

Since we deny Petitioners' petition to suspend and related motions, we need not address the related NEPA issue raised in the motions.[93] Nevertheless, we do so to provide additional clarity regarding the scope of our NEPA responsibilities. NEPA requires us to consider the environmental impacts of major agency actions, such as the issuance of an initial or renewed nuclear power reactor license. In some cases, we have addressed environmental impacts generically.[94] The courts have consistently found generic analyses of the environmental impacts of continued storage and disposal in the context of our reactor licensing proceedings to be acceptable.[95]

Petitioners contend that their requested "safety decision" regarding the feasibility of a repository would constitute a federal action that would require us to prepare a separate NEPA analysis to support our conclusion that spent fuel dis-

---

[88] *See, e.g.,* Corrected comments of "Environmental Organizations on Draft Waste Confidence Generic Environmental Impact Statement and Proposed Waste Confidence Rule and Petition to Revise and Integrate All Safety and Environmental Regulations Related to Spent Fuel Storage and Disposal," at 14, 16 (Jan. 7, 2014) (ADAMS Accession No. ML14024A297).

[89] We responded to the concerns raised by Petitioners in Appendix D of the Continued Storage GEIS. *See, e.g.,* Continued Storage GEIS §§ D.2.1.2, D.2.4.1, and B.2 (discussing the technical feasibility of disposal in a repository).

[90] *Id.* § B.2.

[91] *See generally id.* at B-2 to B-5.

[92] *See, e.g.,* 10 C.F.R. Parts 60, 63, and 72.

[93] Motion at 12-14.

[94] *See, e.g.,* NUREG-1437, Revision 1, Generic Environmental Impact Statement for License Renewal of Nuclear Power Plants — Final Report (June 2013) (ADAMS Accession No. ML13107A023).

[95] *See, e.g., New York,* 681 F.3d at 480 (citing *Baltimore Gas & Electric Co. v. Natural Resources Defense Council, Inc.,* 462 U.S. 87, 100, 103 (1983)) and *Minnesota,* 602 F.2d at 416-17.

posal is technically feasible.[96] Petitioners further assert that this separate analysis was "required by the Court of Appeals in *New York.*"[97] We disagree. We find nothing in the court's decision to support Petitioners' assertion. Nonetheless, any finding we have made, whether express or implied, does not require its own environmental analysis; it is simply a confirmation of what Congress and the courts have previously understood — that we believe it is safe to proceed with reactor licensing because it is ultimately possible to dispose of spent nuclear fuel safely.[98] And of course, each reactor licensing decision will have to be made in light of the full panoply of reasonably foreseeable environmental impacts that can fairly be attributed to the proposed action.[99]

In light of the foregoing, we find that Petitioners have not demonstrated a legal basis for their contention. It follows that Petitioners have not stated a valid contention that satisfies our contention admissibility criteria in 10 C.F.R. § 2.309, nor have they satisfied the criteria to reopen a closed record in 10 C.F.R. § 2.326.[100]

---

[96] Motion at 13.

[97] *Id.* at 14.

[98] In this vein, Petitioners misapprehend our statement in the Continued Storage GEIS that "in this GEIS and Rule, the NRC is not making a safety determination under the Atomic Energy Act . . . to allow for the continued storage of spent fuel. [The Atomic Energy Act] safety determinations would be made as part of individual licensing actions." *See* Motion at 14 n.54 (citing Continued Storage GEIS at D-9). This commitment does not deviate from our long-held view that the [Act] does not require findings regarding spent fuel disposal at the time of reactor or storage facility licensing. We intended only to correct the misimpression that safety findings for the purposes of making final licensing decisions were to be found in our NEPA rulemaking. We therefore noted that these safety findings would be made in future licensing actions as necessary — for example, in the licensing of spent fuel storage facilities after the end of a reactor's license term. The Atomic Energy Act "safety determinations" to which we referred in the Continued Storage GEIS and Rule were not those that Petitioners claim to be required here for spent fuel disposal — they were our well-known determinations that are made as part of final licensing decisions. Continued Storage GEIS at D-9.

[99] Petitioners additionally argue that we must prepare a cost-benefit analysis that considers the "costs of spent fuel storage and disposal" as part of their requested NEPA analysis. Motion to Reopen at 4. In response to comments on the draft Continued Storage GEIS and Rule regarding the cost of continued storage, the Staff added additional information to the Continued Storage GEIS to ensure that NRC decisionmakers, applicants, licensees, and the public would have sufficient information to appropriately consider the costs of continued storage in NEPA analyses for future licensing actions. *See generally* Continued Storage GEIS, Ch. 2. Here, we need not expand upon the disclosure of cost information found in the GEIS. To the extent required by NEPA, the Staff will, as appropriate, consider the cost information contained in Chapter 2 of the GEIS as part of the cost-benefit analyses prepared in conjunction with NEPA reviews for individual licensing proceedings.

[100] Petitioners, Applicants, and the Staff present numerous arguments regarding the procedural propriety of the petition and motions now before us. Because we find that the suspension petition and new contention fail on the merits, and we consider — and take action on — the petition and motions in our supervisory capacity, we need not address these procedural issues. *See, e.g., Callaway,* CLI-11-5, 74 NRC at 158 n.65.

B. Additional Considerations Concerning the Issuance of Licenses

For the reasons discussed above, we do not interpret the Atomic Energy Act to require us to make safety findings regarding the technical feasibility of a repository as a prerequisite to our reactor licensing decisions. We are nonetheless aware of the public's concerns about the safety issues associated with the waste generated by the facilities that we license. For this reason, we stress that our ongoing efforts to ensure adequate protection of the public health and safety are not circumscribed by a narrow conception of what the law requires or a stagnant approach to regulation. Accordingly, we set forth below the considerations that guide our analysis of these issues and our conclusion that licensing nuclear plants will not endanger the public health and safety.

As an initial matter, the disposal question is inextricably linked to the question of the technical feasibility of safe storage pending disposal. As we acknowledged in the Continued Storage GEIS, the time frames we considered, including one that contemplates indefinite storage, depend on the continued technical feasibility of safely storing spent fuel as it ages.[101] Our regulations, including those in 10 C.F.R. Parts 50, 52, and 72, establish stringent safety requirements that apply to the construction and operation of reactor spent fuel pools and independent spent fuel storage installations.[102] Even after the end of a reactor's license term, these storage facilities will continue to be subject to our regulations governing spent fuel storage, which ensure that these safety requirements remain in place for as long as the fuel is stored.[103] For example, 10 C.F.R. § 50.54(bb), which requires licensees to submit for NRC approval their plans to manage spent fuel after the permanent cessation of reactor operation; and 10 C.F.R. Part 50, Appendix A, Criterion 61, which requires that spent fuel storage systems be designed to assure adequate safety under normal and postulated accident conditions, directly relate to the safe storage of spent fuel after a reactor has stopped operating.

Spent fuel can be stored safely in spent fuel pools or independent spent fuel storage installations licensed under the Atomic Energy Act. Indeed, we recently concluded in our Continued Storage rulemaking that the indefinite storage of spent fuel in dry casks, if it becomes necessary, is technically feasible.[104] As reflected in the Continued Storage GEIS, several characteristics of dry cask storage systems

[101] Continued Storage GEIS §§ B.2 and B.3.

[102] *See, e.g., id.* § D.2.4.1, at D-28 to D-32.

[103] *Id.*

[104] In accordance with the direction of the court of appeals, we analyzed a scenario where a repository never becomes available. *New York,* 681 F.3d at 479. As part of this analysis, we determined that it is technically feasible to store spent fuel indefinitely, should it become necessary to do so. Continued Storage GEIS § B.3.

ensure that these systems can safely store spent fuel; among others, these systems are massive, passive, and inherently robust.[105]

Further, our regulatory process is dynamic: we continue to revise and refine our regulatory regime as our technical knowledge and experience grow.[106] Thus, we rely both upon our ability to ensure that licensees conform to existing regulations and upon our comprehensive regulatory scheme that takes into account the length of time during which, and the conditions under which, the storage of spent fuel will occur. For example, in our waste confidence proceedings, we assessed the technical feasibility of geologic disposal, along with the continued storage of spent fuel pending the availability of a repository. As early as 1990, however, we recognized that the length of the continued storage period could be significantly longer than the specific time periods originally reflected in the Temporary Storage Rule.[107] But we did not examine the safety or environmental consequences of storing fuel for longer time frames because we assumed that the Department of Energy would have a deep geologic repository available within those time frames.[108] We revisited this assumption as a consequence of the remand in *New York v. NRC,* and we now have analyzed the impacts of spent fuel storage over much longer time frames.[109] We expect that our regulatory process will not be static and will continue to evolve in the future.

Disposal in a deep geologic repository remains the option that Congress has selected for addressing the problem of spent nuclear fuel, and we have neither a mandate nor a reason to question this determination. For the reasons stated in the Continued Storage GEIS, we believe that a geologic repository is technically feasible and that, with sufficient political and societal commitment, a repository can become available within 25-35 years.[110] But we have no crystal ball. We recognize, as we did in 1977, that the hazards associated with spent fuel could become acute at some distant time. We also recognize, as we must, that our statutory mission only confers upon us the authority to license, and not to construct, a permanent repository.[111] Thus, our statutory obligation to ensure

---

[105] *Id.*

[106] *See, e.g.,* Final Rule: "License and Certificate of Compliance Terms," 76 Fed. Reg. 8873 (Feb. 16, 2011) (extending the maximum possible length of licenses issued under 10 C.F.R. Part 72 from 20 years to 40 years).

[107] In our 1990 Waste Confidence Decision, we noted that "[a]lthough the Commission does not dispute the statement that dry spent fuel storage is safe and environmentally acceptable for a period of 100 years, the Commission does not find it necessary to make that specific finding in this proceeding." 1990 Waste Confidence Decision, 55 Fed. Reg. at 38,473.

[108] *See id.* at 38,482.

[109] *See, e.g.,* Continued Storage GEIS, Chs. 4 and 5.

[110] *Id.* § B.2.

[111] The Nuclear Waste Policy Act assigned the responsibility for constructing and operating a *(Continued)*

the adequate protection of public health and safety encompasses an ongoing responsibility to regulate the continued storage of spent fuel, with or without a repository. Our long history with these issues (including our ability to adapt our regulatory processes based upon changing circumstances) continues to support our conclusion that safe, permanent disposal of spent nuclear fuel is technically feasible and that spent fuel can be safely stored until a repository is available, or indefinitely should such storage become necessary.

Congress has entrusted this agency to ensure adequate protection of public health and safety by granting us the authority to condition licenses and to enforce our regulations. In our view, licensing production and utilization facilities now and relying upon our overall regulatory regime to address both ongoing safe storage and the construction of a repository in the future do not constitute an abdication of our statutory obligations. Rather, we understand these actions to be precisely what Congress intended when it both authorized the NRC to issue licenses for nuclear power plants and granted the agency broad regulatory and enforcement authority to protect the public health and safety and common defense and security.

III. CONCLUSION

In light of these considerations, and in light of our determination that the Atomic Energy Act does not require us to make the "waste confidence safety finding" that Petitioners propose, we decline to suspend final licensing decisions in the captioned proceedings. We therefore *deny* Petitioners' suspension requests and *deny* Petitioners' associated motions for leave to file new contentions and to reopen the record.

IT IS SO ORDERED.

For the Commission

ANNETTE L. VIETTI-COOK
Secretary of the Commission

Dated at Rockville, Maryland,
this 26th day of February 2015.

---

repository to the Department of Energy, not the NRC. *See, e.g.,* Nuclear Waste Policy Act of 1982, § 114, 42 U.S.C. § 10134 (2012).

U.S.NRC
United States Nuclear Regulatory Commission
Protecting People and the Environment

NUREG-2157
Volume 1

Generic Environmental Impact Statement for Continued Storage of Spent Nuclear Fuel

Final Report

Office of Nuclear Material Safety and Safeguards

AVAILABILITY OF REFERENCE MATERIALS IN NRC PUBLICATIONS

NRC Reference Material

As of November 1999, you may electronically access NUREG-series publications and other NRC records at NRC's Public Electronic Reading Room at http://www.nrc.gov/reading-rm.html. Publicly released records include, to name a few, NUREG-series publications; *Federal Register* notices; applicant, licensee, and vendor documents and correspondence; NRC correspondence and internal memoranda; bulletins and information notices; inspection and investigative reports; licensee event reports; and Commission papers and their attachments.

NRC publications in the NUREG series, NRC regulations, and Title 10, "Energy," in the *Code of Federal Regulations* may also be purchased from one of these two sources.

1. The Superintendent of Documents
   U.S. Government Printing Office
   Mail Stop SSOP
   Washington, DC 20402–0001
   Internet: bookstore.gpo.gov
   Telephone: 202-512-1800
   Fax: 202-512-2250
2. The National Technical Information Service
   Springfield, VA 22161–0002
   www.ntis.gov
   1–800–553–6847 or, locally, 703–605–6000

A single copy of each NRC draft report for comment is available free, to the extent of supply, upon written request as follows:
Address: U.S. Nuclear Regulatory Commission
Office of Administration
Publications Branch
Washington, DC 20555-0001
E-mail: DISTRIBUTION.RESOURCE@NRC.GOV
Facsimile: 301–415–2289

Some publications in the NUREG series that are posted at NRC's Web site address http://www.nrc.gov/reading-rm/doc-collections/nuregs are updated periodically and may differ from the last printed version. Although references to material found on a Web site bear the date the material was accessed, the material available on the date cited may subsequently be removed from the site.

Non-NRC Reference Material

Documents available from public and special technical libraries include all open literature items, such as books, journal articles, transactions, *Federal Register* notices, Federal and State legislation, and congressional reports. Such documents as theses, dissertations, foreign reports and translations, and non-NRC conference proceedings may be purchased from their sponsoring organization.

Copies of industry codes and standards used in a substantive manner in the NRC regulatory process are maintained at—
The NRC Technical Library
Two White Flint North
11545 Rockville Pike
Rockville, MD 20852–2738

These standards are available in the library for reference use by the public. Codes and standards are usually copyrighted and may be purchased from the originating organization or, if they are American National Standards, from—
American National Standards Institute
11 West 42nd Street
New York, NY 10036–8002
www.ansi.org
212–642–4900

Legally binding regulatory requirements are stated only in laws; NRC regulations; licenses, including technical specifications; or orders, not in NUREG-series publications. The views expressed in contractor-prepared publications in this series are not necessarily those of the NRC.

The NUREG series comprises (1) technical and administrative reports and books prepared by the staff (NUREG–XXXX) or agency contractors (NUREG/CR–XXXX), (2) proceedings of conferences (NUREG/CP–XXXX), (3) reports resulting from international agreements (NUREG/IA–XXXX), (4) brochures (NUREG/BR–XXXX), and (5) compilations of legal decisions and orders of the Commission and Atomic and Safety Licensing Boards and of Directors' decisions under Section 2.206 of NRC's regulations (NUREG–0750).

U.S.NRC
United States Nuclear Regulatory Commission
Protecting People and the Environment

NUREG-2157
Volume 1

Generic Environmental Impact Statement for Continued Storage of Spent Nuclear Fuel

Final Report

Manuscript Completed: August 2014
Date Published: September 2014

Waste Confidence Directorate
Office of Nuclear Material Safety and Safeguards
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

1.0 Introduction

Since the inception of commercial nuclear power, the United States has worked to find a disposal solution for spent nuclear fuel (spent fuel) generated by commercial nuclear power reactors. In the late 1970s, the U.S. Nuclear Regulatory Commission (NRC) reexamined an underlying assumption used in licensing reactors to that time—*that a repository could be secured for the ultimate disposal of spent fuel generated by nuclear reactors, and that spent fuel could be safely stored in the interim*. This analysis was called the Waste Confidence proceeding.

This *Generic Environmental Impact Statement for Continued Storage of Spent Nuclear Fuel* (GEIS) addresses the environmental impacts of continuing to store spent fuel at a reactor site or at an away-from-reactor storage facility, after the end of the licensed life for operations of a reactor[1] until final disposition in a geologic repository ("continued storage"), historically addressed as part of the NRC's waste confidence proceeding. This GEIS has been prepared to fulfill the Commission's obligations under the National Environmental Policy Act of 1969, as amended (NEPA) and NRC regulations implementing NEPA in Title 10 of the *Code of Federal Regulations* (CFR) Part 51.

1.1 History of Waste Confidence

The first Waste Confidence rulemaking began in the late 1970s in response to two significant legal proceedings. In 1977, the Commission denied a petition for rulemaking filed by the Natural Resources Defense Council (NRDC) that asked the NRC to determine whether radioactive wastes generated in nuclear power reactors can be disposed of without undue risk to public health and safety and to refrain from granting pending or future requests for reactor operating licenses until the NRC made a determination regarding disposal. The Commission stated in its denial that, as a matter of policy, it "... would not continue to license reactors if it did not have reasonable confidence that the wastes can and will in due course be disposed of safely" (42 FR 34391). The Commission's denial of the NRDC petition was affirmed upon judicial review (*NRDC v. NRC*). Since that time, the Federal government has adopted deep geologic disposal as the national solution for spent fuel disposal (Nuclear Waste Policy Act of 1982). Recently, the U.S. Department of Energy (DOE) reaffirmed the Federal government's commitment to the ultimate disposal of spent fuel and predicted that a repository would be available by 2048 (DOE 2013).

[1] As used in the GEIS, the term "licensed life for operation" of a reactor is the period running to the end of the operating license term for a reactor, which may include the term of a revised or renewed license.

At about the same time the Commission denied the NRDC petition, the State of Minnesota and the New England Coalition on Nuclear Pollution challenged license amendments that permitted expansion of the capacity of spent fuel storage pools at two nuclear power plants, Vermont Yankee and Prairie Island. In 1979, the Court of Appeals for the District of Columbia (D.C.) Circuit, in *Minnesota v. NRC*, remanded to the Commission the question of whether an offsite storage or disposal solution would be available for the spent fuel at the two facilities at the expiration of their licenses—at that time scheduled for 2007 and 2009—and, if not, whether the spent fuel could be stored safely at those reactor sites until an offsite solution was available.

In 1979, the NRC initiated a generic rulemaking that stemmed from these challenges and the Court of Appeals' remand in *Minnesota v. NRC*. The Waste Confidence rulemaking generically assessed whether the Commission could have reasonable assurance that spent fuel produced by nuclear power plants "…can be safely disposed of…when such disposal or offsite storage will be available, and…whether radioactive wastes can be safely stored onsite past the expiration of existing facility licenses until offsite disposal or storage is available" (44 FR 61372). On August 31, 1984, the Commission published the Waste Confidence decision (49 FR 34658) (Decision) and a final Rule (49 FR 34688), which codified elements of the decision at 10 CFR 51.23 (Rule) and adopted revisions to 10 CFR Part 50 that established procedures to "…confirm that there will be adequate lead time for whatever actions may be needed at individual reactor sites to assure that the management of spent fuel following the expiration of the reactor operating license will be accomplished in a safe and environmentally acceptable manner" (49 FR 34689). In addition to addressing the NRC's assessment of the issues presented by the Court of Appeals' remand, the Decision provided an environmental assessment (EA) and finding of no significant impact (FONSI) to support the Rule (NRC 1989).

The analysis in 10 CFR 51.23 found that, for at least 30 years beyond the expiration of a reactor's licensed life for operations, no significant environmental impacts would result from storage of spent fuel, and expressed the Commission's reasonable assurance that a repository was likely to be available in the 2007 to 2009 timeframe. The Rule also stated that, as a result of this generic determination, the NRC need not prepare any site-specific environmental analysis in connection with continuing storage when issuing a license or amended license for a new reactor or independent spent fuel storage facility (ISFSI) (10 CFR 51.23(b)).

The first review of the Decision and the Rule occurred in 1989 and 1990. This review resulted in revisions to the Decision and the Rule to reflect revised expectations for the availability of the first repository, and to clarify that the expiration of a reactor's licensed life for operations referred to the full 40-year initial license for operations and a 30-year revised or renewed license. On September 18, 1990, the Commission published the revised Decision (55 FR 38474) and final Rule (55 FR 38472).

The Commission conducted its second review of the Decision and the Rule in 1999 and concluded that experience and developments after 1990 had confirmed the findings and made a comprehensive reevaluation of the Decision and Rule unnecessary. The Commission also stated that it would consider undertaking a comprehensive reevaluation when the pending repository development and regulatory activities had run their course or if significant and pertinent unexpected events occurred that raised substantial doubt about the continuing validity of the Waste Confidence decision (64 FR 68005).

In 2008, the Commission decided to conduct its third review of the Decision and the Rule. This review resulted in revisions to reflect revised expectations for the availability of the first repository and to encompass at least 60 years of continued storage. In December 2010, the Commission published its revised Decision (75 FR 81032) and final Rule (75 FR 81037).

In response to the 2010 rulemaking, the States of New York, New Jersey, Connecticut, and Vermont; several public interest groups; and the Prairie Island Indian Community sought review in the Court of Appeals challenging the Commission's NEPA analysis that supported the Rule. On June 8, 2012, the Court of Appeals ruled that some aspects of the 2010 Waste Confidence rulemaking did not satisfy the NRC's NEPA obligations. The Court of Appeals therefore vacated the Decision and the Rule and remanded the case to the NRC for further proceedings consistent with the Decision (*New York v. NRC*).

The Court of Appeals concluded that the Waste Confidence rulemaking proceeding is a major Federal action necessitating either an environmental impact statement (EIS) or an EA that results in a FONSI. The Court of Appeals identified three deficiencies in the NRC's environmental analysis:

1. Related to the Commission's conclusion that permanent disposal will be available "when necessary," the Court of Appeals held that the Commission needed to evaluate the environmental effects of failing to secure permanent disposal, given the uncertainty about whether a repository would be built.
2. Related to 60 years of continued storage, the Court of Appeals concluded that the Commission had not adequately examined the risk of spent fuel pool leaks in a forward-looking fashion.
3. Also related to continued storage, the Court of Appeals concluded that the Commission had not adequately examined the consequences of potential spent fuel pool fires.

In response to the Court of Appeals' decision, the Commission stated in Commission Order CLI–12–16 that it would not issue reactor or ISFSI licenses dependent upon the Waste Confidence Rule until the Court of Appeals' remand is appropriately addressed (NRC 2012a). This decision is not an indication that the Commission lacks confidence in the availability of an ultimate disposal solution, but rather reflects the Commission's need to develop an analysis that

assesses the environmental impacts of continued storage in a manner addressing the Court of Appeals' remand.[2] The Commission stated, however, that this determination extends only to issuance of the reactor or ISFSI license, and that all licensing reviews and proceedings should continue to move forward. In SRM–COMSECY–12–0016, the Commission directed the NRC to develop a GEIS to support an updated Waste Confidence decision and rule (NRC 2012b).

1.2 Scope of the Generic Environmental Impact Statement

This GEIS analyzes the environmental impacts of continued storage and provides a regulatory basis for the revision to the NRC's Waste Confidence Rule.

The Waste Confidence Rule, originally adopted by the Commission in 1984, satisfies part of the Commission's NEPA obligation to prepare an environmental analysis in the course of a licensing proceeding for a commercial nuclear power reactor or a facility that will store the spent fuel generated by these reactors.

For both power reactor and storage facilities, NEPA requires that the NRC address direct, indirect, and cumulative impacts of its licensing actions. Thus, in issuing a power reactor license, the NRC must analyze the environmental impacts resulting from the generation of spent fuel by the reactor and its continued storage pending ultimate disposal. Likewise, for an ISFSI, the NRC must analyze the impacts of continued storage at the facility until ultimate disposal for the spent fuel is available. The environmental impacts addressed in this GEIS are limited to the environmental impacts of continued storage.

This GEIS considers three possible continued storage timeframes: (1) short-term storage of no more than 60 years after the end of a reactor's licensed life for operations; (2) long-term storage of no more than 160 years after the end of a reactor's licensed life for operations; and (3) indefinite storage at a reactor site or at an away-from-reactor ISFSI. The indefinite storage scenario assumes that disposal in a repository never becomes available.

As discussed above, the NRC has analyzed three timeframes that represent various scenarios for the length of continued storage that will be needed before spent fuel is sent to a repository. The first, most likely, timeframe is the short-term timeframe, which analyzes 60 years of continued storage after the end of a reactor's licensed life for operations. As discussed in more detail later in this GEIS and in Appendix B to this GEIS, the NRC believes this is the most likely

[2] "Waste confidence undergirds certain agency licensing decisions, in particular new reactor licensing and reactor license renewal. Because of the recent court ruling striking down our current waste confidence provisions, we are now considering all available options for resolving the waste confidence issue, which could include generic or site-specific NRC actions, or some combination of both. We have not yet determined a course of action. But, in recognition of our duties under the law, we will not issue licenses dependent upon the Waste Confidence Decision or the Temporary Storage Rule until the court's remand is appropriately addressed." (NRC 2012a) at 4 *citations omitted*.

timeframe because the DOE has expressed its intention to provide repository capacity by 2048, which is about 10 years before the end of this timeframe for the oldest spent fuel within the scope of this analysis. Further, international and domestic experience with deep geologic repository programs supports a timeline of 25 to 35 years to provide repository capacity for the disposal of spent fuel. The DOE's prediction of 2048 is in line with this expectation. The NRC acknowledges, however, that the short-term timeframe, although the most likely, is not certain. Accordingly, two additional timeframes also are analyzed in this GEIS. The long-term timeframe considers the environmental impacts of continued storage for a total of 160 years after the end of a reactor's licensed life for operations. Finally, although the NRC considers it highly unlikely, this GEIS includes an analysis of an indefinite timeframe, which assumes that a repository does not become available.

1.3 Purpose of the Generic Environmental Impact Statement

The purpose of the GEIS is twofold:

1. To determine 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
2. To determine whether those impacts can be generically analyzed.

In the draft GEIS, the NRC preliminarily identified the environmental impacts of continued storage and determined that they could be addressed generically. In the process of developing this final GEIS, including considering and responding to the substantial volume of public comments the NRC received in response to the draft GEIS and proposed Rule, the NRC has confirmed that the impacts of continued storage can be generically addressed. Therefore, the GEIS 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.

1.4 Proposed Federal Action

The Federal action is the adoption of a revised Rule, 10 CFR 51.23, which codifies (i.e., adopts into regulation) the analysis in the GEIS of the environmental impacts of continued storage of spent fuel.

Having confirmed that the environmental impacts of continued storage can be analyzed generically, the Commission has decided to codify the GEIS impact determinations in a revised rule, 10 CFR 51.23. The rule states that, because the impacts of continued storage have been generically assessed in this GEIS, NEPA analyses for relevant future reactor and spent fuel storage facility licensing actions will not need to separately consider the environmental impacts of continued storage.

have already been constructed and are operating during reactor operations. Therefore, many of the impacts of at-reactor continued spent fuel storage can be determined by comparing onsite activities that occur during reactor operations to the reduced activities that occur during continued storage. Where appropriate, the environmental impacts during reactor operations are drawn from the License Renewal GEIS (NRC 2013d), which evaluates the impacts of continued reactor operation. In addition, this GEIS uses analyses in EAs prepared for ISFSIs and renewals of those ISFSI licenses.

For the impacts of continued storage at an-away-from-reactor ISFSI (Chapter 5), the NRC evaluated the impacts of an ISFSI of the same size as described in the *Final Environmental Impact Statement for the Construction and Operation of an Independent Spent Nuclear Fuel Storage Installation on the Reservation of the Skull Valley Band of Goshute Indians and Related Transportation Facility in Tooele County, Utah* (NRC 2001). Chapter 5 contains a list of the assumptions used in that analysis. Unlike in Chapter 4, the generic analysis for away-from-reactor storage at an ISFSI includes a general discussion of the construction of the facility. However, the site-specific impacts of the construction and operation of any proposed away-from-reactor ISFSI would be evaluated by NRC as part of that ISFSI's licensing process.

For both the at-reactor and away-from-reactor storage sites, the NRC assumes that the construction, operation, and replacement of a dry transfer system (DTS) facility is necessary at some point to handle the transfer of fuel. The physical characteristics of a DTS, which is based on well-understood technology, are explained in more detail in Chapter 2 (see Section 2.1.4).

The GEIS accounts for the age of storage facilities in the evaluation of impacts. For example, a storage cask that was loaded with spent fuel 40 years prior to the end of the licensed life for reactor operations has already been in service for 40 years at the beginning of the short-term timeframe and is assumed to be replaced at the beginning of the long-term timeframe (40 years of service at the beginning of the short-term timeframe plus 60 years of service over the short-term timeframe results in a total service time of 100 years, which is the assumed replacement period for dry cask storage facilities).

1.8.2 Timeframes Evaluated

The NRC evaluated the environmental impacts of continued storage in three timeframes that begin once the licensed life of the reactor ends—short-term storage, long-term storage, and indefinite storage (see Figure 1-1).

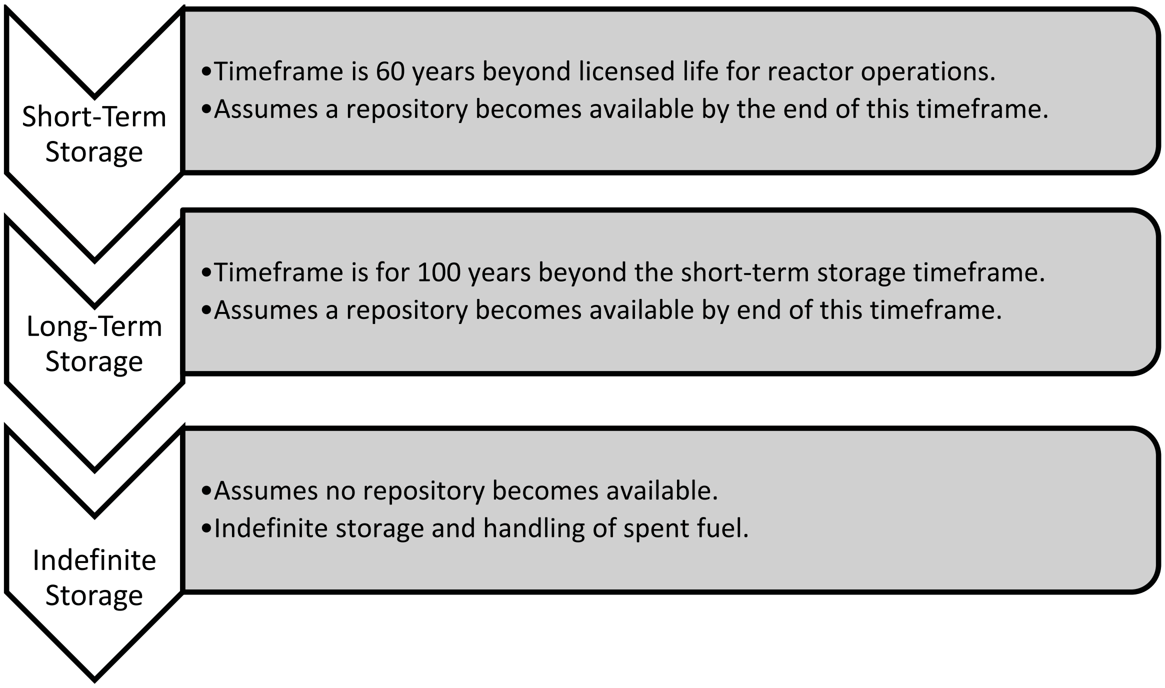


Figure 1-1. Continued Storage Timeframes

The first timeframe—*short-term storage*—lasts for 60 years and begins after the end of a reactor's licensed life for operations. The NRC evaluated the environmental impacts resulting from the following activities that occur during the short-term storage timeframe:

- continued storage of spent fuel in spent fuel pools (at-reactor only) and ISFSIs,
- routine maintenance of at-reactor spent fuel pools and ISFSIs (e.g., maintenance of concrete pads),
- construction and operation of an away-from-reactor ISFSI (including routine maintenance), and
- handling and transfer of spent fuel from spent fuel pools to ISFSIs.

The next timeframe—*long-term storage*—is 100 years and begins immediately after the short-term storage timeframe. The NRC evaluated the environmental impacts resulting from the following activities that occur during long-term storage:

- continued storage of spent fuel in ISFSIs, including routine maintenance,
- one-time replacement of ISFSIs and spent fuel canisters and casks, and
- construction and operation of a DTS (including replacement).

For the long-term storage timeframe, the NRC assumes that all spent fuel has already been moved from the spent fuel pool to dry cask storage by the end of the short-term storage timeframe. The spent fuel pool would be decommissioned within 60 years after permanent cessation of operation, as required by 10 CFR 50.82 or 10 CFR 52.110.

The third timeframe—*indefinite storage*—assumes that a geologic repository does not become available. In this timeframe, at-reactor and away-from-reactor ISFSIs would continue to store spent fuel in dry casks indefinitely. For the evaluation of environmental impacts if no repository becomes available, the following activities are considered:

- continued storage of spent fuel in ISFSIs, including routine maintenance,
- replacement of ISFSIs and spent fuel canisters and casks every 100 years,
- construction and operation of an away-from-reactor ISFSI (including replacement every 100 years), and
- construction and operation of a DTS (including replacement every 100 years).

These activities are the same as those that would occur for long-term storage, but without a repository, they would occur repeatedly.

1.8.3 Analysis Assumptions

To evaluate the potential environmental impacts of continued storage, this GEIS makes several assumptions.

- Although the NRC recognizes that the precise time spent fuel is stored in pools and dry cask storage systems will vary from one reactor to another, this GEIS makes a number of reasonable assumptions regarding the length of time the fuel can be stored in a spent fuel pool and in a dry cask before the fuel needs to be moved or the facility needs to be replaced. With respect to spent fuel pool storage, the NRC assumes that all spent fuel is removed from the spent fuel pool and placed in dry cask storage in an ISFSI no later than 60 years after the end of the reactor's licensed life for operation. With respect to dry cask storage, the NRC assumes that the licensee uses a DTS during long-term and indefinite storage timeframes to move the spent fuel to a new dry cask every 100 years. Similarly, the NRC assumes that the DTS and the ISFSI pad are replaced every 100 years. For an ISFSI that reaches 100 years of age near the end of the short-term storage timeframe, the NRC assumes that the replacement would occur during the long-term storage timeframe.
- Based on its knowledge of and experience with the structure and operation of the various facilities that will provide continued storage, including the normal life of those facilities, the NRC believes that spent fuel pool storage could last for about 60 years beyond the licensed life for operation of the reactor where it is stored, and that each ISFSI will last about 100 years.

- The most reasonably foreseeable assumption is that institutional controls (i.e., the continued regulation of spent fuel) will continue. 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 institutional controls, including but not limited to NRC licensing and regulatory controls, few impacts could be reliably forecast. For the purpose of the analyses in this GEIS, the NRC assumes that regulatory control of radiation safety will remain at the same level of regulatory control as currently exists today. Section B.3.4 of Appendix B provides further discussion regarding institutional controls.
- A DTS will be built at each ISFSI location during the long-term storage timeframe to facilitate spent fuel transfer and handling.
- The NRC assumes a 100-year replacement cycle for spent fuel canisters and casks. This assumption is consistent with assumptions made in the Yucca Mountain Final EIS (DOE 2008).
- The 100-year replacement cycle also assumes replacement of the ISFSI facility and DTS.
- Based on currently available information, the 100-year replacement cycle provides a reasonably conservative assumption for a storage facility that would require replacement at a future point in time. However, this assumption does not mean that dry cask storage systems and facilities *need* to be replaced every 100 years to maintain safe storage.
- 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 (approximately 100 years after spent fuel would have first been transferred from the spent fuel pool into a dry cask storage system, which would occur about 35 years into a reactor’s licensed life for operations).
- The NRC assumes that the land used for the ISFSI pads and DTS would be reclaimed after the facilities are demolished and, therefore, would be used again in the next 100-year replacement cycle. The NRC assumes the initial replacement ISFSI and DTS would be built near the existing facilities. The NRC believes this assumption is reasonable because the characteristics of the previously disturbed land are already known and are suitable for ISFSI and DTS design and construction.
- The NRC assumes that aging management, including routine maintenance activities and programs, occurs between replacements. These “routine” or planned maintenance activities are distinct from the “replacement” of facilities and equipment.
- The spent fuel is moved from the spent fuel pool to dry cask storage within the short-term storage timeframe.
- Under NRC regulations, a nuclear power plant that operates for the term specified in its license is required to complete decommissioning within 60 years after the licensed life for operations in accordance with 10 CFR 50.82 or 52.110. Under these regulations, a plant that permanently ceases operation before the term specified in its operating license is

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.

Module design. The fuel types in these designs range from a mix of uranium-zirconium or uranium-plutonium-zirconium metal alloys to stainless-steel-clad uranium nitride.

These fuel types have not completed fuel qualification testing and are not yet commercially viable technologies. If these technologies should become viable and the NRC is asked to review one or more license applications for a liquid metal fast reactor facility, then the environmental impacts of continued storage of that spent fuel will be considered in individual licensing proceedings unless the NRC updates the GEIS and corresponding rule to include the environmental impacts of storing this type of fuel after a reactor's licensed life for operation.

2.1.2 Onsite Spent Fuel Storage and Handling

As of the end of 2011, the amount of commercial spent fuel in storage at commercial nuclear power plants was an estimated 67,500 MTU. The amount of spent fuel in storage at commercial nuclear power plants is expected to increase at a rate of approximately 2,000 MTU per year (CRS 2012).

Licensees have designed spent fuel pools to temporarily store spent fuel in pools of continuously circulating water that cool the spent fuel assemblies and provide shielding from radiation. When the nuclear power industry designed the current fleet of operating nuclear power plants, it expected that, after a few years, the plant operators would transport spent fuel to one or more reprocessing plants. However, as a result of historic decision-making on reprocessing[8] no commercial spent fuel reprocessing facilities are currently operating or planned in the United States (Copinger et al. 2012).

2.1.2.1 Spent Fuel Pools

Spent fuel pools are designed to store and cool spent fuel following its removal from a reactor. Spent fuel pools are massive and durable structures constructed from reinforced-concrete walls and slabs that vary between 0.7 and 3 m (2 and 10 ft) thick. Typically, spent fuel pools are at least 12 m (40 ft) deep, allowing the spent fuel to be covered by at least 6 m (20 ft) of water, which provides adequate shielding from the radiation for anyone near the pool. 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.) (Copinger et al. 2012). Further, all spent fuel pools have either a leak-detection system or administrative controls to monitor the spent fuel pool liner. Typically, leak-detection systems are made up of several individually monitored channels or are designed

[8] In furtherance of anti-proliferation policies, the Federal government declared a moratorium on reprocessing spent fuel in 1976. This moratorium was lifted in 1981, but in 1993, President Clinton issued a policy statement that the United States does not encourage civil use of plutonium, including reprocessing. In 2001, President Bush's National Energy Policy encouraged research into reprocessing technologies. Currently, there is no Federal moratorium on reprocessing.

so that leaked water empties into monitored drains. Leaked water is directed to a sump, liquid radioactive waste treatment system, or other cleanup or collection system.

Reactor designers originally anticipated that spent fuel would be stored for less than 1 year before being shipped to a reprocessing plant for separation of the fissile isotopes. For this reason, currently operating reactors originally had storage capacity for one full core plus one or two additional discharged batches of spent fuel. When the United States abandoned spent fuel reprocessing and spent fuel pools began to fill up, licensees expanded fuel storage capacity by replacing the original storage racks with higher density fuel racks. Licensees achieved the higher density by taking into account in their safety assessments the neutron-absorbing characteristics of the stainless-steel structure of the storage racks and incorporating plates or sheets containing a neutron absorber material for reactivity control (EPRI 1988). As a result, a typical spent fuel pool at a light water reactor can hold the equivalent of about seven reactor core loads, or about 700 MTU (see Appendix G).

On this basis, a typical spent fuel pool has about 700 MTU storage capacity that reaches its licensed capacity limit in about 35 years into 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.

Spent fuel pools are constructed with the reactor, not during continued storage. Therefore, the cost of building a spent fuel pool facility is not included in this GEIS. However, operating the spent fuel pool is a continued storage activity, and those costs are presented in Section 2.2.1.2.

Two events have resulted in changes to NRC requirements for physical security and the safe operation of spent fuel pools. The first was the terrorist attacks on September 11, 2001, after which the NRC ordered all operating nuclear power plants to immediately implement compensatory security measures. In addition, the NRC issued Orders to decommissioning reactor licensees that imposed additional security measures associated with access authorization, fitness for duty, and behavior observation. In 2009, the NRC completed a rulemaking that codified generally applicable security requirements for operating power plants (74 FR 13926).

Second, in response to the March 11, 2011 severe earthquake and subsequent tsunami that resulted in extensive damage to the six nuclear power reactors at Japan's Fukushima Dai-ichi site, the NRC established a task force of senior agency experts (Near-Term Task Force). On July 12, 2011, the Near-Term Task Force issued its report, which concluded that there was no imminent risk from continued operation and licensing activities (NRC 2011a). Based on its analysis, the Near-Term Task Force made 12 overarching recommendations for changes to ensure the continued safety of U.S. nuclear power plants.

Several of these recommendations addressed spent fuel pool integrity and assurance of adequate makeup water in the event of a serious accident. In response to the Near-Term Task Force's recommendations, the NRC issued multiple Orders and a request for information to all of its operating power reactor licensees and holders of construction permits in active or deferred status on March 12, 2012. The Orders addressed (1) mitigating strategies for beyond-design basis external events and (2) reliable spent fuel pool instrumentation. In addition, the NRC issued the request for information to assist the agency in reevaluating seismic and flooding hazards at operating reactor sites and determining 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. The NRC will use the information collected to determine whether to update the design basis and systems, structures, and components important to safety, including spent fuel pools. However, because the NRC has not yet decided whether any license needs to be modified, suspended, or revoked, for purposes of analysis in this GEIS, the NRC assumes that the related existing regulatory framework remains unchanged. Further, the NRC has initiated a rulemaking to address a condition known as station blackout, a situation that involves the loss of all onsite and offsite alternating current power at a nuclear power plant. The advance notice of proposed rulemaking was published on March 20, 2012 (77 FR 16175), and the draft regulatory basis was published on April 10, 2013 (78 FR 21275). Among other issues being considered as part of the rulemaking, the NRC is evaluating whether to require additional equipment (e.g., backup power supplies and instrumentation) to ensure the safety of spent fuel pools. Current information regarding the status of this proposed rule can be found on the regulations.gov website (www.regulations.gov) under Docket ID NRC-2011-0299.

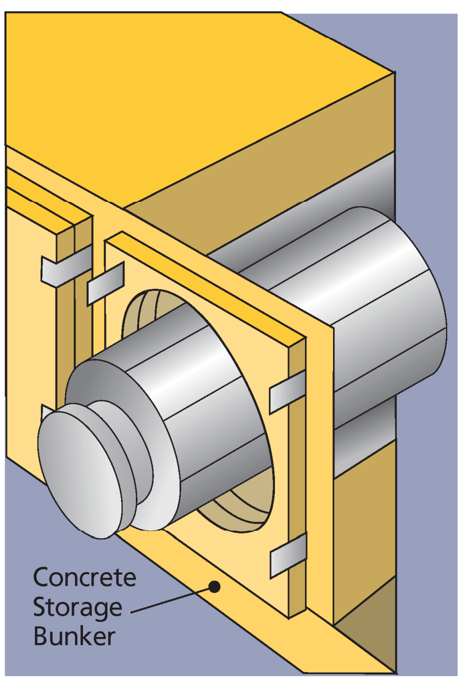
2.1.2.2 At-Reactor Independent Spent Fuel Storage Installations

Spent fuel pools, as discussed above, have limited capacity to store a reactor's spent fuel. As noted, a typical spent fuel pool has a storage capacity of about 700 MTU that reaches its licensed capacity limit about 35 years into licensed life for operation of a reactor. At that point, the licensee needs a dry cask storage system to store older fuel that has cooled sufficiently and can be removed safely from the pool. These dry cask storage systems are located in ISFSIs at reactor sites and are licensed by the NRC. Dry cask storage systems shield people and the environment from radiation and keeps the spent fuel dry and nonreactive (NRC 2013b).

There are many different dry cask storage systems, but most fall into two main categories based on how they are loaded. The first is the bare fuel, or direct-load, casks, in which spent fuel is loaded directly into a basket that is integrated into the cask. Bare fuel casks, which tend to be all metal construction, are generally bolted closed. The second is a canister-based system in which spent fuel is loaded into a basket inside a cylinder called a canister. The canister is usually loaded while inside a transfer cask, then welded and transferred vertically into either a concrete or metal storage overpack or horizontally into a concrete storage module (e.g., NUHOMS) (Hanson et al. 2012). Typical dry cask storage systems are shown in Figure 2-1.

At some nuclear reactors across the country, spent fuel is kept onsite, typically above ground, in systems basically similar to the ones shown here.

1 Once the spent fuel has sufficiently cooled, it is loaded into special canisters that are designed to hold nuclear fuel assemblies. Water and air are removed. The canister is filled with inert gas, welded shut, and rigorously tested for leaks. It is then placed in a cask for storage or transportation. The NRC has approved the storage of up to 40 PWR assemblies and up to 68 BWR assemblies in each canister. The dry casks are then loaded onto concrete pads.



2 The canisters can also be stored in above ground concrete bunkers, each of which is about the size of a one-car garage.

Figure 2-1. Dry Storage of Spent Fuel (Source: NRC 2013b)

Dry cask storage systems are licensed by the NRC for storage only or for storage and transportation. Storage-only casks are not certified for transportation under 10 CFR Part 71, "Packaging and Transportation of Radioactive Material." Casks and canisters licensed for both storage and transportation are generally referred to as dual-purpose casks and dual-purpose canisters. Some vendors refer to their dual-purpose casks or canisters as "multipurpose" canisters, which implies that it would be suitable for storage, transportation, and disposal. However, in the absence of a repository program, there are no specifications for disposal canisters and, therefore, no dual-purpose casks or canisters have been certified as multipurpose (Hanson et al. 2012).

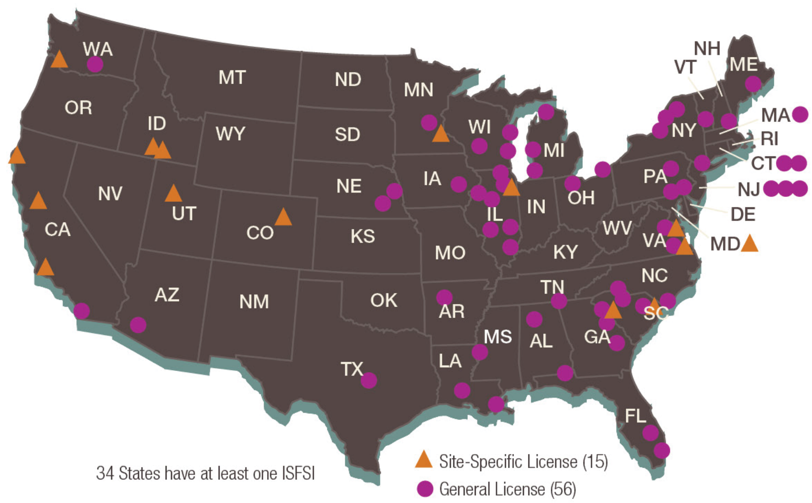
As of June 2014, there were operational ISFSIs at 64 sites. One operational ISFSI, at the GEH Morris site, is a wet storage facility. The remaining ISFSIs store spent fuel in over 1,900 loaded dry casks. Two licenses have been issued for ISFSIs, the PFS facility and the Idaho Spent Fuel Facility, neither of which have been constructed. Figure 2-2 shows the locations of U.S. ISFSIs. Information on ISFSIs is presented in Appendix G of this GEIS.

The NRC authorizes construction and operation of ISFSIs by general and specific licenses. A general license is created by regulation and confers the right upon the general licensee to proceed with the licensed activity without further review or approval by the NRC. A specific license, by contrast, requires an application to perform the licensed activity and NRC review and approval to proceed with the licensed activity.

As these concepts apply to ISFSIs, every nuclear power reactor licensee, by virtue of the general license in 10 CFR Part 72, Subpart K, is authorized to store spent fuel in casks whose design has been approved by the NRC. Licensees must evaluate the safety of using the approved casks at the ISFSI for site-specific conditions, including man-made and natural hazards, and must conform to all requirements under Subpart K for use of the approved design. In addition, licensees must review their programs for operating the reactor (e.g., physical security, radiation protection, and emergency planning) to determine if those programs are affected by use of the casks and, if so, to seek approval from the NRC for any necessary changes to those programs.

Further, a reactor licensee can seek a specific license to construct and operate an ISFSI, which requires NRC's review of the safety, environmental, and physical security aspects of the proposed facility and the licensee's financial qualifications. If the NRC concludes the proposed ISFSI meets licensing criteria, then the NRC grants the specific license. This license contains various conditions (e.g., leak testing and monitoring) and specifies the quantity and type of material the licensee is authorized to store at the site. A specific license runs for a term of up to 40 years and may be renewed in accordance with all applicable requirements.

Licensed and Operating Independent Spent Fuel Storage Installations by State



ALABAMA
- ● Browns Ferry
- ● Farley

ARIZONA
- ● Palo Verde

ARKANSAS
- ● Arkansas Nuclear

CALIFORNIA
- ▲ Diablo Canyon
- ▲ Rancho Seco
- ● San Onofre
- ▲ Humboldt Bay

COLORADO
- ▲ Fort St. Vrain

CONNECTICUT
- ● Haddam Neck
- ● Millstone

FLORIDA
- ● St. Lucie
- ● Turkey Point

GEORGIA
- ● Hatch
- ● Vogtle

IDAHO
- ▲ DOE: TMI-2 (Fuel Debris)
- ▲ Idaho Spent Fuel Facility

ILLINOIS
- ● Braidwood
- ● Byron
- ▲ GE Morris (Wet)
- ● Dresden
- ● La Salle
- ● Quad Cities
- ● Zion

IOWA
- ● Duane Arnold

LOUISIANA
- ● River Bend
- ● Waterford

MAINE
- ● Maine Yankee

MARYLAND
- ▲ Calvert Cliffs

MASSACHUSETTS
- ● Yankee Rowe

MICHIGAN
- ● Big Rock Point
- ● Palisades
- ● Cook

MINNESOTA
- ● Monticello
- ▲ Prairie Island

MISSISSIPPI
- ● Grand Gulf

NEBRASKA
- ● Cooper
- ● Ft. Calhoun

NEW HAMPSHIRE
- ● Seabrook

NEW JERSEY
- ● Hope Creek
- ● Salem
- ● Oyster Creek

NEW YORK
- ● Indian Point
- ● FitzPatrick
- ● Ginna
- ● Nine Mile Point

NORTH CAROLINA
- ● Brunswick
- ● McGuire

OHIO
- ● Davis-Besse
- ● Perry

OREGON
- ▲ Trojan

PENNSYLVANIA
- ● Limerick
- ● Susquehanna
- ● Peach Bottom

SOUTH CAROLINA
- ●▲ Oconee
- ●▲ Robinson
- ● Catawba

TENNESSEE
- ● Sequoyah

TEXAS
- ● Comanche Peak

UTAH
- ▲ Private Fuel Storage

VERMONT
- ● Vermont Yankee

VIRGINIA
- ●▲ Surry
- ●▲ North Anna

WASHINGTON
- ● Columbia

WISCONSIN
- ● Point Beach
- ● Kewaunee
- ● LaCrosse

Data as of June 1, 2014

Figure 2-2. Licensed/Operating ISFSIs by State (Source: NRC 2014)

As described in more detail in Section 2.2.1, nuclear power plant licensees will undertake major decommissioning activities during the 60 years following permanent cessation of reactor operations. During major decommissioning activities, the licensees will transfer spent fuel from spent fuel pools to either an at-reactor or away-from-reactor ISFSI. When decommissioning of the reactor and related facilities is completed and the at-reactor ISFSI is the only spent fuel storage structure left onsite, the facility is referred to as an “ISFSI-only site.” Existing ISFSI-only sites include Big Rock Point, Haddam Neck, Fort St. Vrain, Maine Yankee, Rancho Seco, Trojan, and Yankee Rowe.

The NRC requires licensees to develop spent fuel management plans that include specific consideration of a plan for removal of spent fuel stored under a general license, and spent fuel management before decommissioning systems and components needed for moving, unloading, and shipping spent fuel (10 CFR 50.54(bb) and 72.218).[9]

Construction of a replacement at-reactor ISFSI is a continued storage activity in the long-term and indefinite timeframes. The Electric Power Research Institute (EPRI) developed a formula for estimating the cost to design, license, and construct a dry cask storage facility (EPRI 2012). EPRI’s cost estimate is based in part on the number of casks at the facility. For cost estimates in this GEIS, the NRC uses the EPRI value of 10 MTU per cask (EPRI 2009), which translates to 160 casks for a 1,600 MTU at-reactor ISFSI. Based on EPRI’s formula and its 2012 data, a single 1,600 MTU storage capacity facility costs $107,000,000 ($107M) to design, license, and construct.

Following the terrorist attacks on September 11, 2001, the NRC issued Orders to ISFSI licensees to require certain compensatory measures. For example, on May 23, 2002, the NRC issued an Order to the GEH Morris wet storage ISFSI (NRC 2002b). On October 16, 2002, the NRC also issued Orders to specifically licensed and generally licensed dry storage ISFSIs (including those with near-term plans to store spent fuel in an ISFSI under a general license). The details of these Orders are withheld from the public for security reasons.

In addition to NRC licensing requirements, licensees may also be subject to individual State requirements. For example, the State of Minnesota Public Utilities Commission requires an applicant to receive a “certificate of need” prior to constructing an ISFSI.

<u>Example of At-Reactor ISFSIs</u>

Dry cask storage systems in use in the United States are summarized in Appendix G. Two common systems are described below.

---

[9] The regulations reference “irradiated-fuel-management plans.” For the purposes of this discussion there is no difference between irradiated fuel and spent fuel.

A common vertical dry cask storage system currently in use in at-reactor ISFSIs is Holtec International's HI-STORM 100. The HI-STORM cylindrical overpack is stored on an ISFSI pad with its longitudinal axis in a vertical orientation and could contain, for example, a single Holtec MPC-32 multipurpose canister, which can hold up to 32 PWR fuel assemblies. Compatible canisters are also available for BWR spent fuel. As a result, dry storage of the entire 1,600 MTU of spent fuel generated by a typical reactor, assuming all spent fuel is eventually transferred from the spent fuel pool, would require about 100 casks. Each storage cask is about 3.4 m (11 ft) wide and 6.1 m (20 ft) tall. The layout of casks on an ISFSI pad is guided by operational considerations at each site. However, a nominal layout involves casks separated by about 4.5 m (15 ft). Therefore, a typical ISFSI pad with 100 casks located inside a protected area common to the power plant, and arranged as 10 rows of 10 casks each, would cover about 46 × 46 m (150 × 150 ft) for a total area of about 0.2 ha (0.5 ac) (Holtec 2000). For purposes of analysis in this GEIS, the NRC assumes that an ISFSI of sufficient size to hold all spent fuel generated by a reactor is constructed during the reactor's licensed life for operation.

A common horizontal dry cask storage system currently in use in at-reactor ISFSIs is available from Transnuclear, Inc., a wholly-owned subsidiary of AREVA North America. The NUHOMS horizontal cask system uses dry shielded canisters that are placed in concrete horizontal storage modules (HSMs). Among the compatible NRC-approved canister designs is the NUHOMS-61BT dry shielded canister. This canister, for example, can hold 61 BWR fuel assemblies. Canisters are also available for PWR spent fuel. For a BWR, the HSM is about 6.0 m (20 ft) long, 4.6 m (15 ft) high and 2.9 m (9.7 ft) wide. As a result, dry storage of 1,600 MTU of spent fuel generated by a generic BWR, assuming all spent fuel is eventually transferred from the spent fuel pool to an at-reactor ISFSI, would require about 150 HSMs. If HSMs were installed in rows and placed back-to-back in 2 × 10 arrays, an ISFSI with 150 HSMs would require about 7 double module rows and a single module row of 10 HSMs. Allowing for a 6-m- (20-ft-) wide concrete approach slab on the entrance side of each HSM, a 150 HSM ISFSI site would be about 60 m (200 ft) wide and 220 m (720 ft) long. Therefore, the total area of the horizontal ISFSI, including the protected area, would be about 1.3 ha (3.6 ac).

2.1.3 Away-from-Reactor ISFSIs

Existing away-from-reactor ISFSIs include the GEH Morris wet storage facility in Morris, Illinois, and the DOE's Three Mile Island, Unit 2 Fuel Debris ISFSI at the Idaho National Engineering Laboratory. Further, the NRC has issued a license to PFS for an away-from-reactor ISFSI, which would have been located on the reservation of the Skull Valley Band of Goshute Indians (NRC 2004b).

A future away-from-reactor ISFSI could accept spent fuel from one or more nuclear power plants. For purposes of this GEIS, the NRC assumes that the nuclear power industry could develop an away-from-reactor ISFSI that would store up to 40,000 MTU of spent fuel from various nuclear power plant sites using existing technologies.

> ***Start-up costs*** *include the design, engineering, and licensing costs associated with constructing a storage facility.*
>
> ***Storage facility capital costs*** *include the construction, material, and equipment costs for the storage pads and the various support buildings.*
>
> ***Transportation capital costs*** *include infrastructure (e.g., rail spurs), transportation equipment (e.g., rail locomotives and cars), and transportation casks and associated equipment.*

Construction of away-from-reactor ISFSIs is a continued storage activity for the short-term, long-term, and indefinite timeframes. For an away-from-reactor ISFSI, the initial construction cost is different than subsequent replacement construction costs because of transportation. For spent fuel transportation, continued storage only addresses the one-time transfer of spent fuel from the at-reactor ISFSI to an away-from reactor ISFSI. Therefore, transportation capital costs are only included in the initial construction of an away-from-reactor ISFSI. For continued storage, subsequent replacement of an away-from-reactor ISFSI excludes transportation capital costs because the spent fuel is already located at the site. EPRI estimated the costs of constructing a 40,000 MTU ISFSI (EPRI 2009). The EPRI estimate is based in part on the number of casks at the facility. For cost estimates in this GEIS, the NRC uses the EPRI value of 10 MTU per cask (EPRI 2009) which translates to 4,000 casks for a 40,000 MTU away-from-reactor ISFSI. Based on 2009 data from EPRI (EPRI 2009), the NRC estimates initial construction costs for a 40,000 MTU away-from-reactor interim storage facility at $680M, which includes $74.2M for start-up costs, $141M for facility capital costs, and $465M for transportation capital costs. Excluding the transportation capital cost reduces the price for building a replacement away-from-reactor ISFSI at that location (i.e., subsequent replacement construction cost) to $215M. Activity costs associated with transportation are described in GEIS Section 2.2.1.4.

Spent fuel would be moved from operating or decommissioning reactor sites, or ISFSI-only sites, to an away-from-reactor ISFSI or ISFSIs, and then from the away-from-reactor ISFSI to one or more permanent repositories. Aside from the existing GEH Morris wet storage facility, and for the purposes of the analysis in this GEIS, the NRC assumes that, in the future, a portion of the nuclear power industry's spent fuel would be stored in one or more dry cask storage systems at an away-from-reactor ISFSI.

In 2006, the NRC granted a license to PFS, to construct and operate an away-from-reactor ISFSI in Skull Valley, Utah. PFS, a consortium of eight nuclear power utilities, proposed to construct the site on the reservation of the Skull Valley Band of Goshute Indians, about 80 km (50 mi) southwest of Salt Lake City, Utah. The PFS facility was intended for temporary aboveground storage, using the Holtec HI-STORM dual-purpose canister-based cask system, of up to 40,000 MTU of spent fuel from U.S. commercial nuclear power plants. PFS proposed to build the ISFSI on a 330-ha (820-ac) site leased from the Skull Valley Band of Goshute Indians. The site would be located in the northwest corner of the reservation approximately 6 km (3.5 mi)

from the Skull Valley Band's village. The proposed PFS ISFSI has not been constructed. Despite the PFS facility not having been constructed, issuance of the PFS license supports the assumption in this GEIS that an away-from-reactor ISFSI is feasible and that the NRC can license an away-from-reactor storage facility. Thus, the NRC's analysis of construction, operation, and decommissioning activities and impacts for an away-from-reactor ISFSI in NUREG–1714 are reflected in this GEIS (NRC 2001).

Consolidated Storage

On January 29, 2010, the President of the United States directed the Secretary of Energy to establish a "Blue Ribbon Commission on America's Nuclear Future." The Blue Ribbon Commission was tasked with conducting a comprehensive review of policies for managing the back end of the nuclear fuel cycle and recommending a new strategy. The Blue Ribbon Commission issued its findings and conclusions in January 2012 (BRC 2012). Among the findings and conclusions related to continued storage of spent fuel was a strategy for prompt efforts to develop one or more consolidated storage facilities.

In January 2013, DOE published its response to the Blue Ribbon Commission recommendations titled, *Strategy for the Management and Disposal of Used Nuclear Fuel and High-Level Radioactive Waste* (DOE 2013). This strategy implements a program over the next 10 years that, with congressional authorization, will:

- site, design, construct, license, and begin operation of a pilot interim storage facility by 2021 with an initial focus on accepting spent fuel from shutdown reactor sites,
- advance toward the siting and licensing of a larger interim storage facility to be available by 2025 with sufficient capacity to provide flexibility in the waste-management system and allow for acceptance of enough spent fuel to reduce expected government liabilities, and
- make demonstrable progress on the siting and characterization of repository sites to facilitate the availability of a geologic repository by 2048.

The Federal government's support for interim storage supports the NRC's decision to consider this type of facility as one of the reasonably foreseeable interim solutions for spent fuel storage pending ultimate disposal at a repository.

2.1.4 Dry Transfer System

Although there are no dry transfer systems (DTSs) at U.S. nuclear power plant sites today, the potential need for a DTS, or facility with equivalent capability, to enable retrieval of spent fuel from dry casks for inspection or repackaging will increase as the duration and quantity of fuel in dry storage increases. A DTS would enhance management of spent fuel inspection and repackaging at all ISFSI sites and provide additional flexibility at all dry storage sites by enabling

repackaging without the need to return the spent fuel to a pool. A DTS would also help reduce risks associated with unplanned events or unforeseen conditions and facilitate storage reconfiguration to meet future storage, transport, or disposal requirements (Carlsen and Raap 2012).

Several DTS designs and related concepts have been put forward over the past few decades. Among these designs is a design developed by Transnuclear, Inc. in the early 1990s under a cooperative agreement between DOE and EPRI. Although the conceptual design was based on transferring spent fuel from a 30-ton 4-assembly source cask to a 125-ton receiving cask, the DTS could be adapted to be suitable for any two casks (Carlsen and Raap 2012).

On September 30, 1996, the DOE submitted to the NRC for review a topical safety analysis report on the Transnuclear-EPRI DTS design (DOE 1996). In November 2000, the NRC issued an assessment report in which it found the DTS concept has merit. The NRC's assessment was based on the DTS meeting the applicable requirements of 10 CFR Part 72 for spent fuel storage and handling and 10 CFR Part 20 for radiation protection. However, the DOE has not yet requested a Part 72 license for the DTS (NRC 2000).

Construction of a DTS is considered a continued storage activity in the long-term and indefinite timeframes. Based on EPRI data (EPRI 1995), the NRC estimates a construction cost of $8.58M for the development of a DTS to handle bare spent fuel that could accommodate repackaging, as needed, to replace casks. The NRC assumed that estimated construction costs for the DTS are the same for both the at-reactor and away-from-reactor facilities.

The reference DTS considered in this GEIS is a two-level concrete and steel structure with an attached single-level weather-resistant preengineered steel building. The concrete and steel structure provides both confinement and shielding during fuel transfer operations. The DTS was designed to enable loading of one receiving cask in 10 24-hour days and unloading one source cask in one 24-hour day.

The key facility parameters and characteristics described in the September 30, 1996, topical safety analysis report are summarized below.

The reference DTS is a reinforced-concrete rectangular box structure with internal floor dimensions of about 8 × 5.5 m (26 × 18 ft) and about 14 m (47 ft) tall. The system also includes an attached, prefabricated, aluminum Butler-type building referred to as the preparation area with dimensions of about 11.6 x 7.6 m (38 × 25 ft) wide and 11.6 m (38 ft) tall. The basemat for the facility measures 14.9 × 21.9 m (49 × 72 ft), and the security zone would be about 76 × 91 m (250 × 300 ft) (i.e., less than 0.7 ha [2 ac]).

As shown in Figure 2-3, the preparation area is located at ground level of the DTS. The lower access area is next to the preparation area and directly below the transfer confinement area.

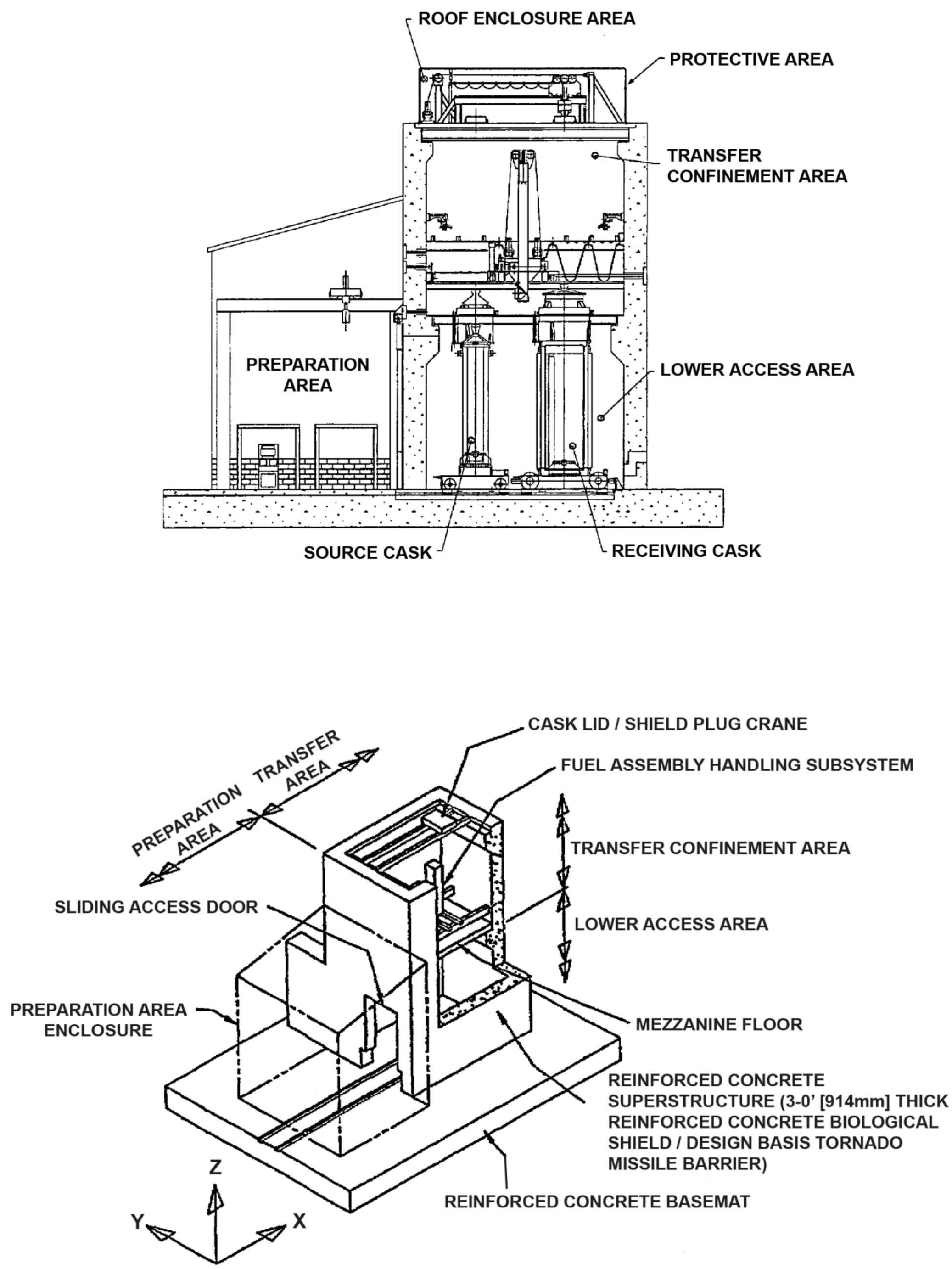


Figure 2-3. Conceptual Sketches of a Dry Transfer System (DOE 1996)

The lower access area provides shielding, confinement, and positioning for the open source and receiving casks during spent fuel transfers. An 18- to 23-cm (7- to 9-in.)-thick steel sliding door separates the lower access area from the preparation area. The transfer confinement area is the upper level of the DTS, directly above the lower access area. The transfer confinement area provides the physical confinement boundary and radiation shielding between spent fuel and the environment.

Transnuclear-EPRI found that radioactive waste generation from dry transfer activities could not be readily quantified, as it depends strongly on reactor-specific conditions, primarily the crud levels on the fuel assemblies. Table 6.1-1 of the topical safety analysis report (DOE 1996) showed the expected waste sources, including decontamination wastes, spalled material in a crud catcher, and prefilters and high-efficiency particulate air filters used in the heating ventilation and air conditioning system. Other wastes considered included mechanical lubricants and precipitation runoff. The DTS does not rely on water-supply lines. Water is brought to the facility in bottles and used for general purpose cleaning only.

The reference DTS, if licensed, would operate under the radiological protection requirements of 10 CFR Part 20, "Standards for Protection against Radiation." Occupational doses for various tasks performed in the DTS are provided in Table 7.4-1 of the topical safety analysis report (DOE 1996). Total estimated occupational doses from loading a single cask are about 0.5 person-rem.

Maximum offsite doses reported in Table 7.6-1 of the topical safety analysis report were estimated to range from 44 mrem per year at 100 m to 2 mrem per year at 500 m.

As with other facilities licensed under 10 CFR Part 72, the design events identified in ANSI/ANS 57.9 (ANSI/ANS 1992) form the basis for the accident analyses performed for the DTS. The bounding accident results for a distance of 100 m are a stuck fuel assembly (47 mrem) and a loss-of-confinement barrier (721 mrem).

This GEIS considers the environmental impacts of constructing a reference DTS to provide a complete picture of the environmental impacts of continued storage. This GEIS does not license or approve construction or operation of a DTS. A separate licensing action would be necessary before a licensee may construct and operate a site-specific DTS.

For the purposes of analysis in this GEIS, the NRC relies primarily on the facility description of the Transnuclear-EPRI DTS described above. However, for some impact assessments in this GEIS, the NRC has drawn from the *Environmental Impact Statement for the Proposed Idaho Spent Fuel Facility at the Idaho National Engineering and Environmental Laboratory in Butte County, Idaho* (NRC 2004b). The NRC licensed the Idaho Spent Fuel Facility in November 2004, but DOE has not constructed the facility. However, the proposed facility has the capability to handle bare spent fuel for the purposes of repackaging and storing spent fuel from

Peach Bottom Unit 1; the Shippingport Atomic Power Station; and various training, research, and isotope reactors built by General Atomics. Because the Idaho Spent Fuel Facility, like the DTS, includes design features that allow bare fuel-handling operations to repackage spent fuel from DOE transfer casks to new storage containers, the NRC has concluded that some environmental impacts of the facility would be comparable to those of a DTS.

2.2 Generic Activity Descriptions

As described in Chapter 1, this GEIS analyzes environmental impacts of the continued storage of spent fuel in terms of three storage timeframes: short-term, long-term, and indefinite storage. As described below, the activities at spent fuel storage facilities during the short-term timeframe coincide with nuclear power plant decommissioning activities. By the beginning of the long-term timeframe, reactor licensees will have removed all spent fuel from the spent fuel pool and decommissioned all remaining nuclear power plant structures. At that point, all spent fuel will be stored in either an at-reactor or away-from-reactor ISFSI. During the long-term storage timeframe, the NRC has conservatively assumed for the purpose of analysis in this GEIS that the need will arise for the transfer of spent fuel assemblies from aged dry cask storage systems to newer systems of the same or newer design. In addition, the NRC assumes that storage pads and modules would need to be replaced periodically. Section 1.8.2 identifies the continued storage activities for which the NRC evaluated the environmental impacts in this GEIS. This section provides the costs for those activities, as well as costs for transporting spent fuel to an away-from-reactor ISFSI during continued storage; the environmental impacts of transporting spent fuel to an away-from-reactor ISFSI are analyzed in Chapter 5.

2.2.1 Short-Term Storage Activities

As depicted in the generic timeline in Figure 2-4, after about 35 years of operation at low fuel burnups, or about 46 years of high-burnup operation, the spent fuel pool at a typical reactor reaches capacity and spent fuel must be removed from the pool to ensure full core offload capability. The inventory of spent fuel that exceeds spent fuel pool capacity may be transferred to dry cask storage at an at-reactor or away-from-reactor ISFSI. This GEIS focuses on the activities and impacts associated with continued storage in a spent fuel pool and dry cask. This section explains the activities that occur during short-term storage:

- decommissioning of the plant systems, structures, and components not required for continued storage of spent fuel,
- routine maintenance of the pool and ISFSI, and
- transfer of spent fuel from the pool to the at-reactor or away-from-reactor ISFSI.

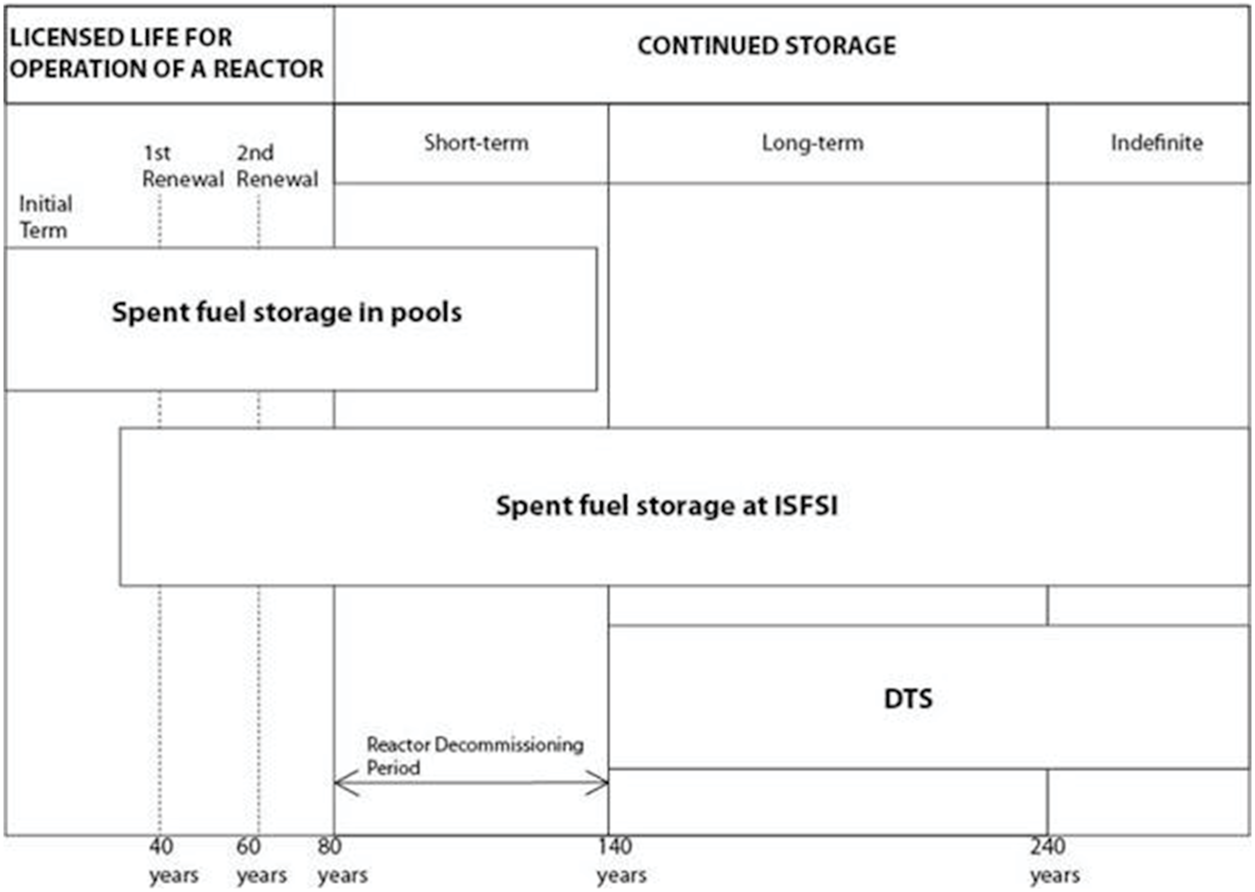


Figure 2-4. Continued Storage Timeline

2.2.1.1 Decommissioning Activities during Short-Term Storage

A number of activities occur after a reactor licensee declares permanent cessation of operations. These activities are divided into three phases: (1) initial activities; (2) major decommissioning and storage activities; and (3) license-termination activities. The initial activities include the licensee's certification to the NRC within 30 days of the decision or requirement to permanently cease operations. This is followed by certification of permanent fuel removal from the reactor. Within 2 years of permanent shutdown, the licensee is required to submit to the NRC a post-shutdown decommissioning activities report that includes a description of planned decommissioning activities along with a schedule, an estimate of expected costs, and a discussion that provides the reasons for concluding that previously issued environmental impact statements bound the site-specific decommissioning activities (NRC 2013c).

Licensees may choose from three decommissioning options: DECON, SAFSTOR, and ENTOMB:

> DECON: The equipment, structures, and portions of the facility and site that contain radioactive contaminants are removed or decontaminated to a level that permits termination of the license shortly after cessation of operations.
>
> SAFSTOR: The facility is placed in a safe, stable condition and maintained in that state until it is subsequently decontaminated and dismantled to levels that permit license termination. During SAFSTOR, a facility is left intact, but the fuel is removed from the reactor vessel and radioactive liquids are drained from systems and components and then processed. Radioactive decay occurs during the SAFSTOR period, which reduces the levels of radioactivity in and on the material and, potentially, the quantity of material that must be disposed of during decontamination and dismantlement.
>
> ENTOMB: ENTOMB involves encasing radioactive structures, systems, and components within 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[10] (NRC 2013c). The NRC has previously considered a range of likely ENTOMB scenarios. For all scenarios considered, spent fuel was removed from the spent fuel pool prior to entombment (NRC 2002a). While the nuclear power industry has expressed interest in maintaining the option for ENTOMB, no licensees have committed to using it (NRC 2002c).

The choice of decommissioning option is left to the licensee, but decommissioning must conform to the NRC's regulations. This choice is communicated to the NRC and the public in the post-shutdown decommissioning activities report. In addition, the licensee may choose to combine the DECON and SAFSTOR options. For example, after power operations cease at a facility, a licensee could use a short storage period for planning purposes, followed by removal of large components (such as the steam generators, pressurizer, and reactor vessel internals), place the facility in storage for 30 years, and eventually finish the decontamination and dismantlement process (NRC 2013c).

If a licensee needs to change the decommissioning schedules or activities identified in the post-shutdown decommissioning activity report, or if the decommissioning costs increase significantly, 10 CFR 50.82(a)(7) and 52.110(g) require the licensee to notify the NRC in writing

[10] Because most power reactors will have radionuclides in concentrations exceeding the limits for unrestricted use even after 100 years, this option will generally not be feasible (NRC 2013c).

and send a copy to the affected States. The NRC uses the post-shutdown decommissioning activity report and any written notification of changes to manage decommissioning oversight activities.

Decommissioning will be completed within 60 years of permanent cessation of operations in accordance with the license-termination requirements for power reactors in 10 CFR 50.82(a)(3) and 52.110(c). Completion of decommissioning beyond 60 years will be approved by the Commission only when necessary to protect public health and safety. Factors that will be considered by the Commission include unavailability of waste disposal capacity and other site-specific factors, including the presence of other nuclear facilities at the site. Given this regulatory framework, it may be reasonably assumed that each nuclear power plant, including its onsite spent fuel pool, will be decommissioned within 60 years of permanent cessation of operations.

Licensees may begin major decommissioning activities 90 days after the NRC has received the post-shutdown decommissioning activities report. The term “major decommissioning activity” is defined in 10 CFR 50.2 and means, for a nuclear power reactor facility, 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 low-level waste as defined in 10 CFR 61.55. Finally, once decommissioning is completed, and any spent fuel stored by the licensee is removed from the site, a licensee may apply to the NRC to terminate its Part 50 or Part 52 license.[11] A licensee is required by 10 CFR 50.82(a)(9) or 52.110(i)(1) to submit to the NRC a license-termination plan as a supplement to its final safety analysis report at least 2 years prior to the expected termination of the license as scheduled in the post-shutdown decommissioning activities report.

Decommissioning activities are not a part of continued storage. Therefore, decommissioning costs are not included in this GEIS.

2.2.1.2 Activities in Spent Fuel Pools

Spent fuel pools are cooled by continuously circulating water that cools the spent fuel assemblies and provides shielding from radiation. During the short-term storage timeframe, the pools will be used to store fuel until a licensee decides to remove the spent fuel as part of implementing the selected decommissioning option. Beyond the short-term storage timeframe, the NRC assumes that all of the spent fuel has been transferred to a dry cask storage system in an at-reactor or away-from-reactor ISFSI, which is consistent with current practice.

[11] A licensee may terminate its Part 50 or Part 52 license earlier if the remaining spent fuel is stored under a specific license issued under 10 CFR Part 72.

materials for landscaping and site construction from local sources. Commercial mining or quarrying operations are not allowed within nuclear power plant boundaries (NRC 2013a).

3.6 Surface-Water Quality and Use

This section describes the surface-water use and quality that could be affected by the continued storage of spent fuel in spent fuel pools and at-reactor ISFSIs.

Because nuclear reactor operations rely predominantly on water for cooling, most nuclear power plant sites are located near reliable sources of water. These sources are often surface waterbodies such as rivers, lakes, oceans, bays, and reservoirs and other man-made impoundments (NRC 2013a). The single exception is the Palo Verde Nuclear Generating Station in Arizona, which uses treated municipal wastewater for cooling water. Of the sites in the United States that contain NRC-licensed nuclear power plants, 32 are located near rivers, 22 near lakes and reservoirs, 5 near oceans, and 5 near estuaries and bays. These waterbodies form part of the affected environment for storage of spent fuel in spent fuel pools and at-reactor ISFSIs. Local drainage features at and near nuclear power plant sites, such as creeks and small streams, provide avenues for surface-water movement and interaction with surface waterbodies. Depending on regional precipitation regimes, local topography, and drainage patterns, operation of spent fuel pools and at-reactor ISFSIs may affect the availability and quality of these nearby surface-water resources.

Provisions of the Clean Water Act regulate the discharge of pollutants into waters of the United States. Discharges of cooling water and other plant wastewaters are monitored through the National Pollutant Discharge Elimination System (NPDES) program administered by the EPA, or, where delegated, individual States. An NPDES permit is developed with two levels of controls: (1) technology-based limits and (2) water quality-based limits. The technology-based limits applicable to nuclear power-generating plants are in 40 CFR Part 423. NPDES permit terms may not exceed 5 years (unless administratively continued), and the applicant must reapply at least 180 days prior to the permit expiration date. The NPDES permit contains requirements that limit the flow rates and pollutant concentrations that may be discharged at permitted outfalls. Biocides and other contaminants in discharged cooling waters are governed by NPDES permit restrictions to reduce the potential for toxic effects on nontargeted organisms (e.g., native mussels and fish). NPDES permits impose temperature limits for effluents (which may vary by season) and/or a maximum temperature increase above the ambient water temperature (referred to as “delta-T,” which also may vary by season). Other aspects of the permit may include the compliance measuring location and restrictions against plant shutdowns during winter to avoid drastic temperature changes in surface waterbodies. The permit also may include biological monitoring parameters that are primarily associated with the discharge of cooling water. The intake of cooling water from waters of the United States is regulated under Clean Water Act Section 316(b), and the thermal component of any effluent discharges from

power-generating plants may be regulated by either the applicable State water quality standard or by Clean Water Act Section 316(a).

Wastewater discharge is also covered through NPDES permitting, and it includes biochemical monitoring parameters. Conditions of discharge for each plant are specified in its NPDES permit issued by the State or EPA. Most plants have a stormwater management plan, with the parameter limits of the stormwater outfalls included in the NPDES permit. Plants also may have a spill prevention, control, and countermeasures plan that provides information on potential liquid spill hazards and the appropriate absorbent materials to use if a spill occurs.

In an effort to minimize or eliminate impacts to the water quality of receiving waterbodies, best management practices are typically included as conditions within NPDES permits. Best management practices are measures used to control the adverse stormwater-related effects of land disturbance and development. They include structural devices designed to remove pollutants, reduce runoff rates and volumes, and protect aquatic habitats. Best management practices also include nonstructural or administrative approaches, such as training to educate staff on the proper handling and disposal of potential pollutants.

After cessation of reactor operations at the nuclear power plant sites, water use would be reduced to spent fuel pool cooling, radiation protection for workers, maintenance, human consumption, and personal hygiene.

3.7 Groundwater Quality and Use

This section describes the groundwater use and quality that could be affected by the continued storage of spent fuel in spent fuel pools and at-reactor ISFSIs.

Groundwater, which has been used as a water supply source throughout recorded history, is found in the voids of unconsolidated geologic materials (e.g., sand and gravel), in fractures of consolidated rocks (e.g., sedimentary, metamorphic, igneous, and volcanic rocks), and in conduits/channels of carbonates (e.g., limestone and dolomites). Where groundwater can be found in the subsurface depends on the geologic history of an area. The quantity and quality of groundwater for domestic uses depends on site-specific conditions. Anthropogenic impacts may affect groundwater quality, but those impacts also are site-specific. Both unconfined and confined aquifers that can provide a potential water supply source for domestic use may exist beneath a nuclear power plant site. The type of aquifers and their properties at nuclear power plant sites are site-specific and can vary considerably.

In the eastern United States, most nuclear power plant sites are located in two large regional groundwater provinces: (1) the first is composed of the Atlantic and Eastern Gulf coastal plain, the Southeastern coastal plain, and the Gulf of Mexico coastal plain; and (2) the second is composed of the Central Glaciated and the Central Nonglaciated plains (Back et al. 1988). The

Nuclear power plant sites must be located near waterbodies that are large enough to adequately meet the demands of a plant's cooling systems. At-reactor ISFSIs are generally located near nuclear power plants, and nuclear power plant sites are usually located near marine and estuarine coastal areas, on the Great Lakes, or along major rivers or reservoirs. Several power plants are sited near small streams (e.g., the V.C. Summer plant in South Carolina and the Clinton plant in Illinois), and initial construction activities included impounding the streams to create cooling ponds or reservoirs.

To establish the affected environment for this analysis, aquatic resources are described in terms of aquatic habitats (freshwater rivers, reservoirs, lakes, and coastal estuarine and marine systems) and aquatic biota (fish, macroinvertebrates, zooplankton, phytoplankton and macrophytes, other aquatic vertebrates and invertebrates, and aquatic vegetation).

3.9.1 Aquatic Habitats

A wide range of aquatic habitats occur in the vicinity of U.S. nuclear power plant sites due to differences in geographies, physical conditions (e.g., substrate type, temperature, turbidity, and light penetration), chemical conditions (e.g., dissolved oxygen levels and nutrient concentrations), biological interactions (e.g., consumption of various algal and invertebrate species that provide habitats, such as seagrass or shellfish beds), seasonal influences (including climate change), and man-made modifications. The interactions of these factors often define the specific type of aquatic habitats and communities within a particular area. Three main aquatic ecosystem types occur near nuclear power plant sites: freshwater, estuarine, and marine ecosystems.

3.9.1.1 Freshwater Systems

Freshwater systems are generally classified into two groups based on the degree of water movement. Lentic systems are waterbodies with standing or slow-flowing water, such as ponds, lakes, reservoirs, and some canals. During warmer months, the upper and lower depths will stratify or become two layers that have different temperatures, oxygen content, and nutrient content. Lotic habitats, on the other hand, feature moving water and include natural rivers and streams and some artificial waterways. Most lotic habitats do not stratify (Morrow and Fischenich 2000). Some freshwater aquatic species may occur in both lentic and lotic habitats. However, many species are adapted to the physical, chemical, and ecological characteristics of one system or the other and the overall ecological communities present within these aquatic ecosystem types differ for different regions of the country (NRC 2013a).

A number of major rivers provide cooling water for nuclear power plant sites. The geographic area, gradient of the river bed, substrate, temperature, dissolved oxygen concentration, depth, light penetration, velocity of the current, and source of nutrients and organic matter at the base of the food chain will largely determine species composition and ecological conditions within

riverine environments. In some instances, nuclear power plants that use rivers for cooling are located on sections of rivers that have been impounded, creating reservoirs. Impoundment of a river can alter ecological communities occurring in a given waterbody by blocking movement of aquatic organisms, changing flow and temperature characteristics, adding chemical pollutants, and introducing non-native species. Fish species in numerous reservoirs are often stocked and managed to support local recreational fisheries (NRC 2013a).

Littoral, pelagic, and profundal habitat zones are all found within lentic systems and are classified on the basis of water depth and light penetration in the water. Littoral habitats refer to nearshore shallower waters where sufficient light reaches the bottom to enable rooted plants to grow. Pelagic habitats include open offshore waters where light intensity is great enough for photosynthesis to occur. Profundal habitats are found in deep-water areas where light penetration is insufficient to support photosynthesis (Armantrout 1998). Unique ecological communities inhabit each zone, reflecting the preferences and tolerances of various aquatic species (NRC 2013a).

Aquatic Ecosystem Types

- ***Freshwater***: Waters that contain a salt concentration or salinity of less than 0.5 parts per thousand (ppt) or 0.05 percent.
  - *Lentic*: Stagnant or slow-flowing fresh water (e.g., lakes and ponds).
  - *Lotic*: Flowing fresh water with a measurable velocity (e.g., rivers and streams).
- ***Estuarine***: Coastal bodies of water, where freshwater merges with marine waters. The waterbodies are often semi-enclosed and have a free connection with marine ecosystems (e.g., bays, inlets, lagoons, and ocean-flooded river valleys). Salinity concentrations fluctuate between 0 and 30 ppt, varying spatially and temporally due to location and tidal activity.
- ***Marine***: Waters that contain a salt concentration of about 30 ppt (e.g., ocean overlying the continental shelf and associated shores).

In the Great Lakes, species diversity and biomass of fish are greater nearshore than in the offshore areas since these areas feature habitats and conditions that are favorable for most species of Great Lakes fish for at least some portion of their life cycle (Edsall and Charlton 1997). Threats to the ecological integrity of the Great Lakes include eutrophication (nutrient enrichment), land-use changes, overfishing, invasive species, and pollution (Beeton 2002). Regulations and best management practices have been implemented to reduce nutrient inputs and control land-use changes, such as shoreline alteration and destruction of wetlands. Invasive species, however, have become a major problem as nonindigenous species gain access to the Great Lakes. The introduction of invasive species can result in changes to native ecological communities (NRC 2013a).

3.9.1.2 Estuarine Ecosystems

Brackish to saltwater estuarine ecosystems occur along the coastlines of the United States. General habitat types found within estuarine ecosystems include the mouths of rivers, tidal streams, shorelines, salt marshes, mangroves, seagrass communities, soft-sediment habitats

and the facility would produce minimal gaseous or liquid effluents, impacts on aquatic resources from the operation of ISFSIs during short-term storage would not have noticeable impacts on aquatic resources.

4.10.1.3 Conclusion

Given that the impacts associated with the operation of spent fuel pools would likely be bounded by the impacts analyzed in the License Renewal GEIS due to the lower withdrawal rates, lower discharge rate, smaller thermal plume, and lower heat content for a spent fuel pool compared to an operating reactor with closed-cycle cooling, the NRC concludes that impacts on aquatic resources from the operation of spent fuel pools during short-term storage would be minimal. In addition, the impacts from operation of at-reactor ISFSIs would be minimal because ISFSIs do not require water for cooling, produce minimal gaseous or liquid effluents, and ground-disturbing activities for ISFSI maintenance would have minimal impacts on aquatic ecology. Therefore the NRC concludes that the potential environmental impacts on aquatic resources would be SMALL during the short-term storage timeframe.

4.10.2 Long-Term Storage

Routine maintenance and monitoring of the ISFSIs would continue during long-term storage. Likewise, the impacts from routine maintenance and monitoring of ISFSIs during the short-term storage timeframe would continue during the long-term storage timeframe and would remain the same.

Due to the relatively small construction footprint of a DTS, a DTS could likely be sited and constructed on land near existing facilities, on previously disturbed ground, and away from sensitive aquatic features. In addition, the replacement DTS and ISFSI facilities could likely be sited on previously disturbed ground away from sensitive aquatic features. For example, the NRC did not identify any significant impacts on aquatic resources from construction of the Humboldt Bay ISFSI in part due to the fact that ground-disturbing activities would be limited to 0.4 ha (1 ac) and the ISFSI was not located near any aquatic features (NRC 2005a). Similarly, the construction footprint for the Diablo Canyon ISFSI was limited to 2 ha (5 ac) and was sited in a previously disturbed area that did not contain any sensitive aquatic features (NRC 2003). In addition, the NRC (2003, 2005a) indicated that controls would be in place to minimize the flow of any site runoff, spillage, and leaks into sensitive aquatic features. For example, stormwater control measures, which would be required to comply with NPDES permitting, would minimize the flow of disturbed soils or other contaminates into aquatic features. The plant operator could also implement best management practices to minimize erosion and sedimentation.

ISFSIs and DTSs do not require water for cooling and produce minimal gaseous or liquid effluents. In addition, replacement ISFSIs and DTSs would be sited on previously disturbed

ground away from sensitive aquatic features. The older ISFSIs and DTSs would be demolished and the land reclaimed. Therefore, the NRC concludes that impacts on aquatic resources during long-term storage would be SMALL.

4.10.3 Indefinite Storage

During indefinite storage, the activities that occur during long-term storage would continue and the ISFSIs and DTSs would be replaced every 100 years. Therefore the impacts that occurred during long-term storage would continue. The NRC concluded in Section 4.10.2 that impacts on aquatic resources would be SMALL because ISFSIs do not require water for cooling and would have minimal impacts on aquatic resources. In addition, replacement of the ISFSIs and DTSs would occur near existing facilities and would be sited on previously disturbed ground away from sensitive aquatic features. The older ISFSIs and DTSs would be demolished and the land reclaimed. Therefore, the NRC concludes that the impacts on aquatic resources from indefinite storage of spent fuel in at-reactor ISFSIs would be SMALL.

4.11 Special Status Species and Habitat

This section describes potential environmental impacts on special status species and their habitats caused by the continued storage of spent fuel in spent fuel pools and at-reactor ISFSIs. Special status species and habitats may include those identified in Section 4.9 for terrestrial resources and Section 4.10 for aquatic resources.

4.11.1 Short-Term Storage

Impacts on Federally listed species, designated critical habitat, essential fish habitat, and other special status species and habitats during short-term storage may occur from spent fuel pool or ISFSI operations.

4.11.1.1 Spent Fuel Pools

Given that Federally listed species, designated critical habitat, essential fish habitat, State-listed species, marine mammals, migratory birds, and bald and golden eagles may be affected by operation of cooling systems for nuclear power plants, special status species and habitats could also be affected by the operation of cooling systems for spent fuel pools during the short-term storage timeframe. Possible impacts on Federally listed species, designated critical habitat, essential fish habitat, State-listed species, marine mammals, migratory birds, and bald and golden eagles would be similar to those described in Sections 4.9.1 and 4.10.1 for terrestrial and aquatic resources.

The Endangered Species Act (ESA) forbids “take” of a listed species, where “take” means to “harass, harm, pursue, hunt, shoot, wound, kill, trap, capture, or collect, or attempt to engage in

Even in rare cases where an independently operating spent fuel pool causes noise impacts that exceed the EPA-recommended threshold for outdoor noise, licensees are usually able to make engineering changes to address the problem. For example, at the Maine Yankee nuclear power plant the licensee set up the pool storage operations to operate independently from the reactor, which was being decommissioned. The fans used as part of the spent pool cooling-system generated noise levels up to 107 dB, which attenuated to 50 dB less than 1.6 km (1 mi) away (NRC 2002b). This noise level exceeded the 55 dB(A) threshold recommended by the EPA for protection against outdoor activity interference and annoyance. Nearby residents complained to the plant staff about the noise level, and the licensee made engineering changes to the fans that were causing the noise and the issue was resolved.

In conclusion, the operation noise levels, duration, and distance between the noise sources and receptors generally do not produce noise impacts noticeable to the surrounding community. In certain cases, such as the Maine Yankee spent fuel pool island, potential noise impacts on receptors closest to the site property line can experience unmitigated noise levels that exceed EPA-recommended noise levels. However, noticeable noise levels are generally not expected and would be limited to the nearest receptors. Therefore, the NRC concludes that the overall impact from noise during short-term storage would be SMALL.

4.13.2 Long-Term Storage

In addition to routine maintenance and monitoring, the NRC assumes that long-term storage would include the construction, operation, and replacement of a DTS and the replacement of the ISFSI. Construction of a DTS would generate higher noise levels than DTS operations. The NRC assumes that DTS construction would take 1–2 years. Construction equipment would be used to grade and level the site, excavate the facility foundation, handle building materials, and build the facility. Construction equipment generates noise levels over 90 dB(A) (at a reference distance of 15 m [50 ft] from the source) (NRC 2002b). At distances greater than about 1.6 km (1 mi), expected maximum noise levels from construction equipment would be reduced to about 55 dB(A), which is the EPA-recommended level for protection in residential areas against outdoor activity interference and annoyance (NRC 2002b).

During operation of the DTS, some activities would be conducted inside the building, which functions as a noise barrier. Spent fuel transfer between the storage pad and the DTS would be infrequent. The NRC expects noise levels from this transfer of spent fuel to be no more than the noise level generated transferring spent fuel from the pool to the dry pad, as described in Section 4.13.1. In addition, some of the reactor and spent fuel pool storage noise sources present during short-term storage (such as the cooling towers and associated equipment) would not be present during long-term storage.

The NRC assumes that the at-reactor ISFSI (i.e., concrete storage casks and pads) and the DTS would be replaced within the 100-year timeframe. Similar to the DTS construction, ISFSI and DTS replacement uses construction equipment, which can generate noise levels over

90 dB(A). The noise levels exceed the EPA-recommended level for protection against outdoor activity interference and annoyance (NRC 2002b). However, distance from the source will eventually reduce the noise level to below the EPA-recommended level for protection against outdoor activity interference and annoyance.

Construction and replacement of the DTS, although temporary and representing a small portion of the overall long-term storage timeframe, would generate noise levels that exceed EPA-recommended noise levels. Operational noise levels would not produce noise impacts noticeable to the surrounding community. For some activities (e.g., replacement of the DTS and ISFSI facilities), potential noise impacts on receptors closest to the site property line can experience unmitigated noise levels that exceed EPA-recommended noise levels. However, these activities are temporary and noticeable noise levels would be limited to the nearest receptors. Therefore, the NRC concludes that the overall impact from noise during long-term storage would be SMALL.

4.13.3 Indefinite Storage

This section describes the noise impacts in the event a repository is not available to accept spent fuel and the spent fuel must be stored indefinitely in ISFSIs. Impacts from indefinite storage would be similar to those described for the long-term storage timeframe. The NRC does not anticipate that indefinite storage in an ISFSI would generate any new or additional noise in comparison with the noise impacts described for the long-term storage timeframe. Therefore, the NRC concludes that the overall impact from noise during indefinite storage would be SMALL.

4.14 Aesthetics

This section describes potential impacts on aesthetic resources caused by continued storage of spent fuel in spent fuel pools and at-reactor ISFSIs.

4.14.1 Short-Term Storage

No changes to nuclear power plant structures will be required for continued operation of the spent fuel pool during continued storage, including routine maintenance and monitoring.

In the License Renewal GEIS, the NRC determined that the aesthetic impacts associated with continued operation of a nuclear power plant, which included the continued operation of the spent fuel pool, were SMALL because the existing visual profiles of nuclear power plants were not expected to change during the license renewal term (NRC 2013a). Therefore, the NRC concludes that the potential impacts from the short-term continued operation of the spent fuel pool would be of minor significance to aesthetic resources.

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

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

systems. 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; therefore, the variation in equipment and fuel characteristics do not present insurmountable challenges or preclude a generic approach to analysis of impacts. In addition, the NRC requires that facilities and equipment are maintained to ensure safety functions and are not compromised. Further, the NRC inspects operating facilities to verify compliance with requirements.

The impacts from accidents, including those involving transfer operations, are evaluated in Sections 4.18, 5.18, and 6.4.17 of the GEIS. Although the consequences of an accident could be high, the impacts were found to be SMALL based on the low likelihood and, therefore, low risk (see Section D.2.35.27 of this appendix for more information). As described in Section 2.1.4 of the GEIS, a DTS would be licensed by NRC under the regulations in 10 CFR Part 72. Therefore, future licensing of site-specific DTSs would undergo thorough NRC safety and environmental reviews that would consider potential accidents and evaluate in detail how each proposed facility operator would maintain safety in transfer operations involving the specific fuel pool, transfer equipment, and type of dry storage system (including canisters and casks) for that facility.

Radiation monitoring is conducted at all NRC-licensed facilities to comply with the radiation protection program requirements in 10 CFR Part 20. Radiation monitoring verifies that licensees are maintaining control of radioactive materials and not exceeding worker and public dose limits. Any planned radioactive effluents from a DTS would be documented in detail during a site-specific licensing of a transfer facility. An applicant for an NRC license would need to demonstrate how applicable standards for worker and public safety would be met by proposed operations (see Section D.2.34.11 of this appendix for more information).

Regarding the availability of land area to accommodate the construction of a DTS or an ISFSI, as described in Section 3.1 of the GEIS, most U.S. power plants are sited on large tracts of land that have available areas where a DTS or ISFSI could be located. Table 3-1 of the GEIS provides a comparison of the small amount of land required for an ISFSI with the total site area at various power plant sites. If a power plant site with limited available land area did not have sufficient land area to construct a DTS or ISFSI then the licensee would have to pursue other options (e.g., arranging for storage at an away-from-reactor storage facility). The impacts of continued storage at an away-from-reactor storage facility were evaluated in Chapter 5 of the GEIS. No changes were made to the GEIS or Rule as a result of these comments.

(163-34-5) (328-7-4) (459-4) (553-14) (619-1-23) (805-14) (919-4-12)

D.2.17.4 – COMMENT: Several commenters stated that NRC has not described how damaged spent fuel transfer operations can be carried out. The commenters believe significant

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.

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

A001
MRR

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
Rosemead, CA 91770

Prepared by:
Energy*Solutions*, LLC
100 Mill Plain Road
Mailbox No. 106
Danbury, CT 06811

Authored By:	Michael S. Williams	September 5, 2014
	Michael S. Williams, Project Manager	Date
Reviewed By:	Barry Sims	September 5, 2014
	Barry S. Sims, Technical Advisor	Date
Approved By	Michael S. Williams	September 5, 2014
	Michael S. Williams, Project Manager	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**

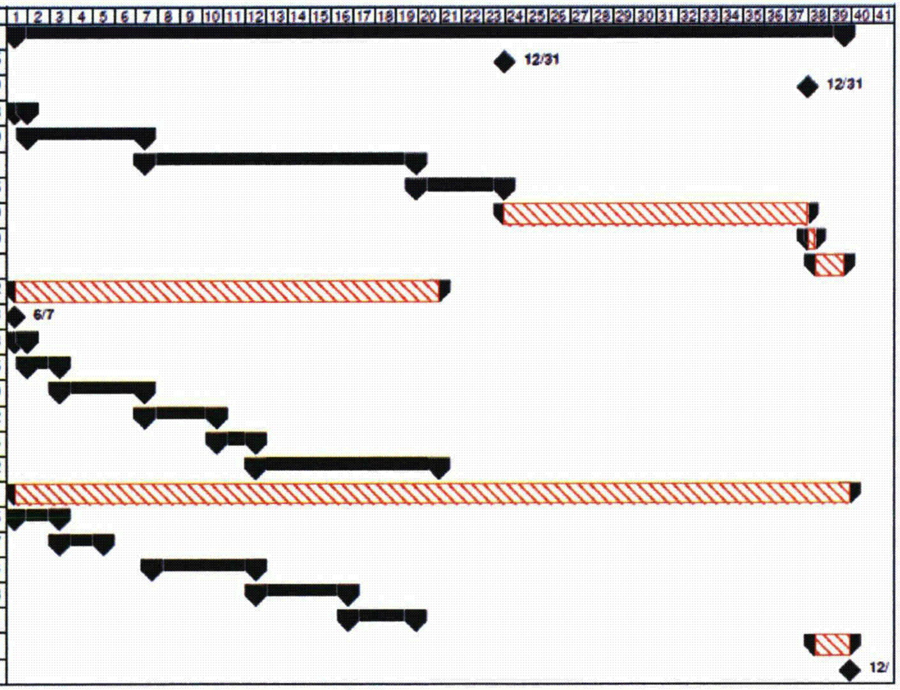
The estimate is based on site-specific plant systems and buildings inventories. These inventories, and Energy*Solutions*' proprietary Unit Cost Factors (UCFs), were used to generate required manhours, activity schedule hours and costs, and waste volume, weight, and classification. Based on the activity 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.

Packaging

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.

Transportation

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.

Class A Disposal Options and Rates

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:

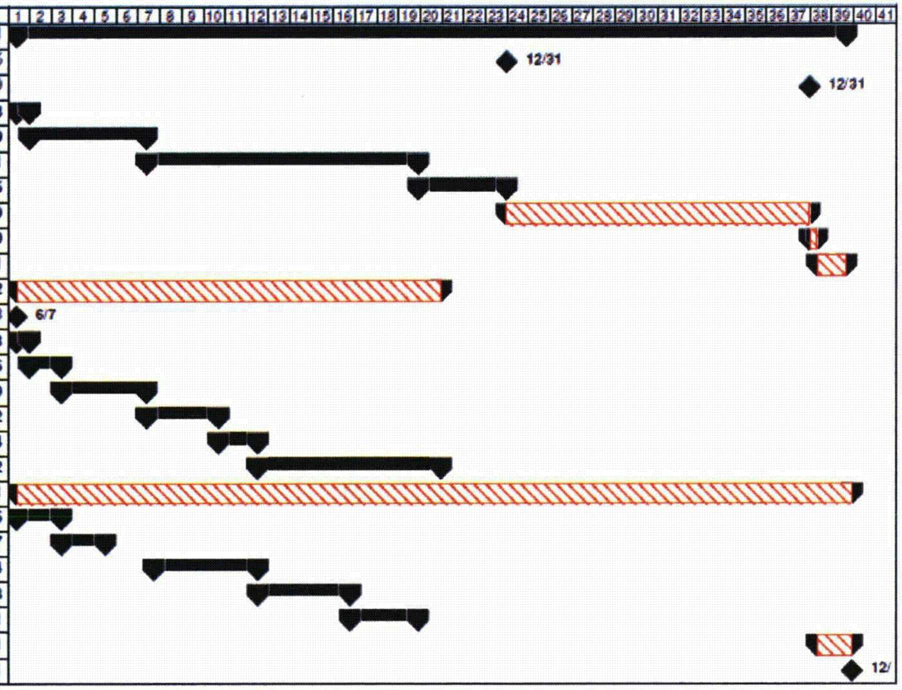
Waste Class	Unit 2	Unit 3	Total
Class A	1,832,961	1,819,680	3,652,641
Class B	7,600	7,600	15,200
Class C	4,095	4,095	8,190
GTCC	941	941	1,882

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



SER 255

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**

7.0 REFERENCES

1. U.S. Nuclear Regulatory Commission, "Domestic Licensing of Production and Utilization Facilities," 10 CFR Part 50, 2008, Available from http://www.nrc.gov/reading-rm/doc-collections/cfr/part050/full-text.html.

2. Atomic Industrial Forum, Inc., "Guidelines for Producing Commercial Nuclear Power Plant Decommissioning Cost Estimates," AIF/NESP-036, May 1986.

3. U.S. Nuclear Regulatory Commission, "Standard Format and Content of Decommissioning Cost Estimates for Nuclear Power Reactors," Regulatory Guide 1.202, February 2005.

4. Federal Register, Vol. 4, "Standard Contract for Disposal of Spent Nuclear Fuel and/or High-Level Radioactive Waste," NRC 10 CFR Part 961 (DOE), January 1, 1999.

5. U.S. Nuclear Regulatory Commission, "Technology, Safety and Costs of Decommissioning a Reference Pressurized Water Reactor Power Station," NUREG/CR-0130, June 1978.

6. U.S. Nuclear Regulatory Commission, "Technology, Safety and Costs of Decommissioning a Reference Boiling Water Reactor Power Station," NUREG/CR-0672, June 1980.

7. Life-of-Plant Disposal Agreement, between Energy*Solutions*, LLC and SCE effective January 1st, 2014.

8. U.S. Nuclear Regulatory Commission, "Multi-Agency Radiation Survey and Site Investigation Manual (MARSSIM)," NUREG-1575, Rev. 1, August 2000.

9. U.S. Department of Energy, "Cost Estimating Guide," DOE G 430.1-1, March 1997.

10. RS Means, "Labor Rates for the Construction Industry," 2014.

11. ABZ, Incorporated, "San Onofre Nuclear Generating Station Units 2 and 3 (SONGS 2/3)", December 14,2012

12. U.S. Department of Energy, "Acceptance Priority Ranking & Annual Capacity Report," DOE/RW-0567, July 2004.

13. Pacific Gas & Electric, "Establishing an Appropriate Contingency Factor for Inclusion in the Decommissioning Revenue Requirements", April 2009

14. Department of the Navy, "Grant of Easement," May 1964.

15. State of California State Lands Commission, "Lease P.R.C. No. 4862.1", dated November 19, 1984

16. State of California, Executive Order D-62-02, September 2002

17. Southern California Edison, "Testimony On the Nuclear Decommissioning Of SONGS 2 & 3 And Palo Verde Before the Public Utilities Commission of the State of California," December 21, 2012.

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 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

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

TABLE OF CONTENTS

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

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