NAC INTERNATIONAL

NAC MULTI-PURPOSE CANISTER (NAC-MPC) SYSTEM

SAFETY EVALUATION REPORT

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INTRODUCTION

This Safety Evaluation Report (SER) documents the review and evaluation of Revision 5 to the Safety Analysis Report (SAR) for the NAC International (NAC) Multi-Purpose Canister (MPC) System. The SAR was submitted by NAC following the format of Regulatory Guide 3.61. This SER primarily uses the Section-level format of NUREG-1536, Standard Review Plan for Dry Cask Storage Systems, with some differences implemented for clarity and consistency.

The review of the SAR addresses the handling and dry storage of spent fuel in a single dry storage cask design, the NAC-MPC. The cask may be used at an Independent Spent Fuel Storage Installation (ISFSI) licensed under Subpart K of 10 CFR Part 72 for a 10 CFR Part 50 licensee.

The staff's assessment is based on whether the applicant meets the applicable requirements of 10 CFR Part 72 for independent storage of spent fuel and of 10 CFR Part 20 for radiation protection. Decommissioning, to the extent that it is treated in the SAR, presumes that, as a bounding case, the NAC-MPC cask is unloaded and decontaminated as necessary before disposition or disposal.

While components of the NAC-MPC system are designed to be used in conjunction with the NAC Storage Transport Cask (NAC-STC) for a dual purpose function, the use or certification of the NAC-STC under 10 CFR Part 71 for the off-site transport of NAC-MPC authorized spent fuel contents is not a subject of this SER. Certification for transportation of the NAC-MPC authorized spent fuel contents occurs upon the completion of a separate staff review to amend the NAC-STC 10 CFR Part 71 Certificate of Compliance (CoC) for transportation.

LIST OF ACRONYMS USED

ACI	American Concrete Institute
ALARA	As Low As Is Reasonably Achievable
ANS	American Nuclear Society
ANSI	American National Standards Institute
ASCE	American Society of Civil Engineers
ASME	American Society of Mechanical Engineers
ASTM	American Society for Testing and Materials
AWS	American Welding Society
CoC	Certificate of Compliance
DBE	Design Basis Earthquake
DCSS	Dry Cask Storage System
DLF	Dynamic Load Factor
ISFSI	Independent Spent Fuel Storage Installation
ISG	Interim Staff Guidance
kW	Kilowatts
MPC	Multi-Purpose Canister
NAC	NAC International, Inc.
NDE	Nondestructive Examination
NRC	Nuclear Regulatory Commission
PT	Liquid Penetrant Examination
PWR	Pressurized Water Reactor
QA	Quality Assurance
RFA	Reconfigured Fuel Assembly

RT	Radiographic Examination
SAR	Safety Analysis Report
SER	Safety Evaluation Report
SSCs	Structures, Systems and Components
STC	Storage Transport Cask
TS	Technical Specifications
TSC	Transportable Storage Canister
UT	Ultrasonic Examination
VCC	Vertical Concrete Cask
VT	Visual Examination

1.0 GENERAL DESCRIPTION

The objective of the review of the general description of the NAC-Multi-Purpose Canister (MPC) system is to ensure that NAC International Inc. (NAC) has provided a non-proprietary description that is adequate to familiarize reviewers and other interested parties with the pertinent features of the system.

1.1 System Description and Operational Features

The NAC-MPC system is a transport compatible dry storage system that uses a stainless steel transportable storage canister (TSC) stored within the central cavity of a vertical concrete cask (VCC). The TSC is designed to be compatible with the NAC Storage Transport Cask (STC) to allow future shipment. The VCC provides radiation shielding and contains internal air flow paths that allow decay heat from the TSC spent fuel contents to be removed by natural air circulation around the canister wall.

The principal components of the NAC-MPC system are the TSC, the VCC, and the transfer cask. The transfer cask is used to move the loaded TSC to and from the VCC, and provides radiation shielding while the TSC is being closed and sealed. The TSC is placed in the VCC by positioning the transfer cask on top of the VCC and subsequently lowering the TSC. Each NAC-MPC system component is assigned a safety classification (Category A, B, or C) in Table 2.3-1 of the Safety Analysis Report (SAR). The component safety classification is based on NUREG/CR-6407, "Classification of Transportation Packaging and Dry Spent Fuel Storage System Components According to Importance to Safety." The principal components of the NAC-MPC system are depicted in Figure 1-1.

The NAC-MPC is designed to store up to 36 Yankee Class pressurized water reactor (PWR) spent fuel assemblies. The spent fuel is loaded into a TSC which contains a stainless steel gridwork referred to as a basket.

1.1.1 Transportable Storage Canister and Baskets

The TSC has an outside diameter of about 71 inches and is about 123-inches long. The weight of the loaded TSC is slightly less than 55,000 lbs. The TSC contains a basket that accommodates up to 36 Yankee Class spent fuel assemblies. The canister assembly consists of a right circular cylindrical shell with a welded bottom plate, a fuel basket, a shield lid, two penetration port covers, and a structural lid. The cylindrical shell, plus the bottom plate and lids, constitutes the confinement boundary. The TSC is shown in Figure 1-2.

The stainless steel fuel basket is a right circular cylinder configuration with 36 fuel tubes laterally supported by a series of stainless steel support disks, which are retained by spacers on 8 radially located tie rods. The square fuel tubes are encased with Boral sheets on all four sides for criticality control. Aluminum heat transfer disks are spaced midway between the support disks and are the primary path for conducting heat from the spent fuel assemblies to the TSC wall.

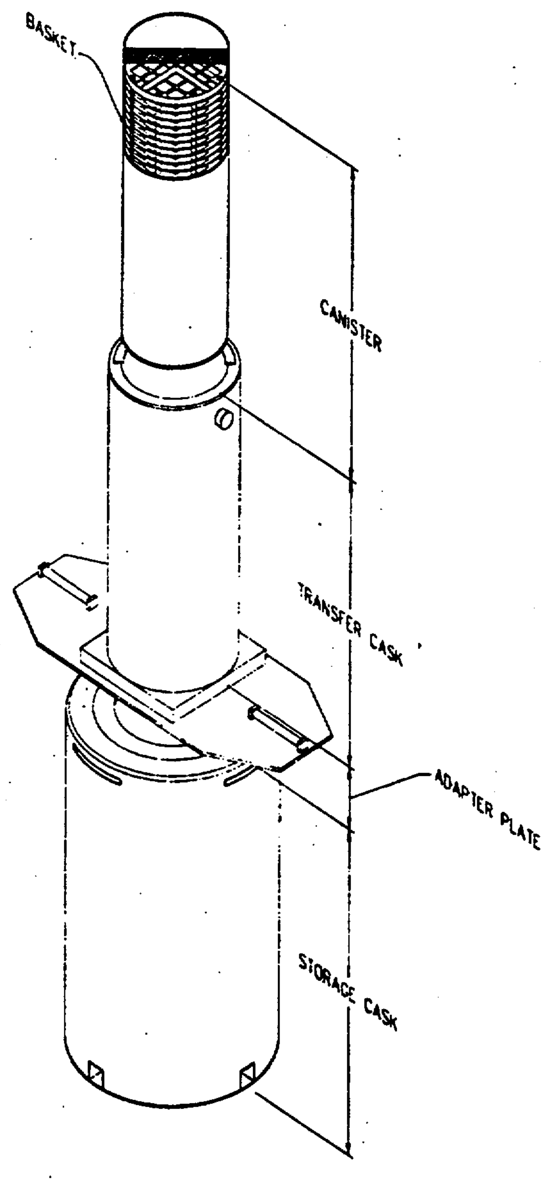

Figure 1-1 Principal Components of the NAC-MPC System

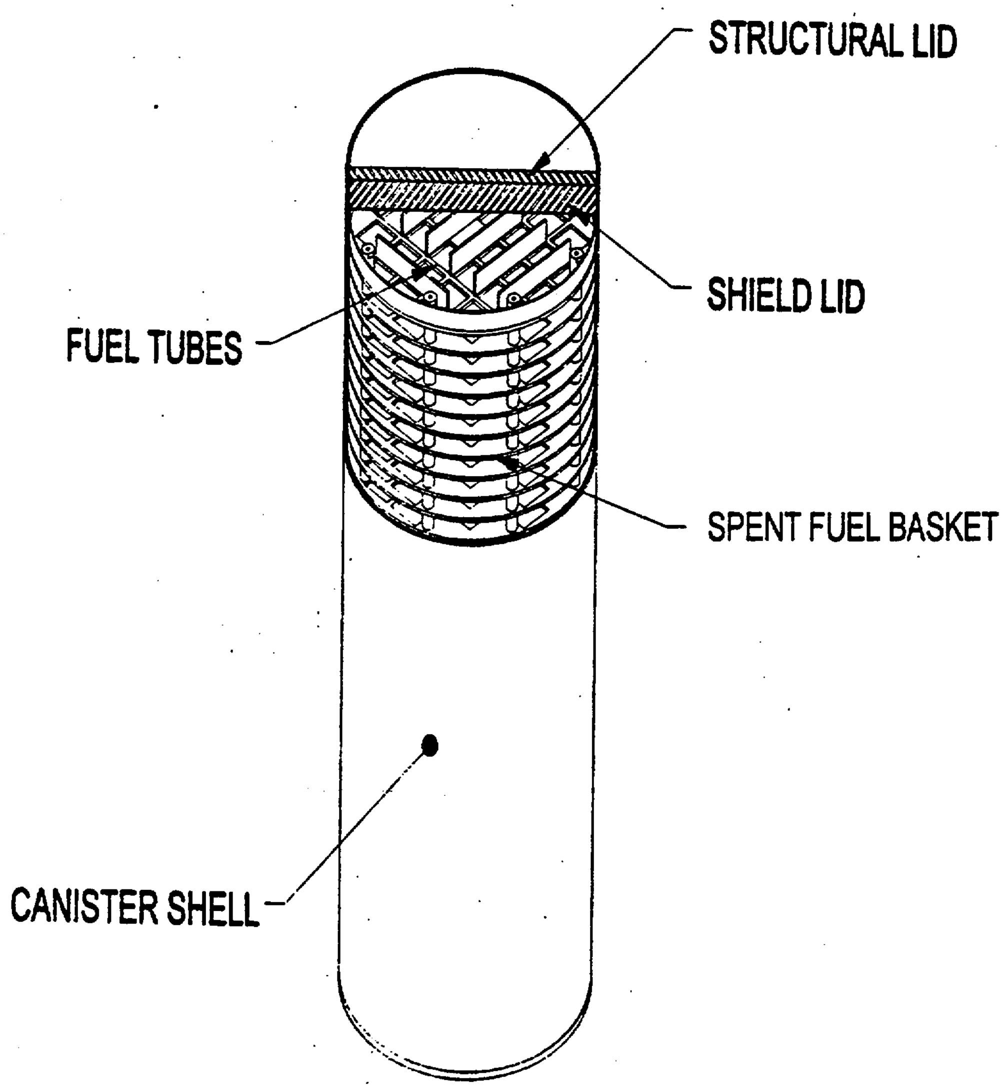

Figure 1-2 Transportable Storage Canister

The TSC assembly is designed to facilitate filling with water and subsequent draining and drying. Vent and drain ports through the shield lid allow the inner cavity to be drained, evacuated, and backfilled with helium to provide an inert atmosphere for long-term storage. After draining, drying, backfilling, and testing operations are completed, port covers are installed and welded to the shield lid to seal the penetration. The designs of the shield and structural lids provide a redundant confinement seal at the top of the canister.

1.1.2 Vertical Concrete Cask

The VCC is the storage overpack for the TSC and is approximately 160 inches in height, has an outside diameter of about 128 inches, and weighs about 155,000 lbs. The VCC side walls consist of about 21 inches of reinforced concrete (Type II Portland cement), with a 3.5 inch carbon steel liner. It provides structural support, shielding, protection from environmental conditions, and natural convection cooling of the TSC during long-term storage. The VCC has an annular air passage to allow the natural circulation of air around the TSC. The air inlet and outlets take non-planar paths to the VCC cavity to minimize radiation streaming. The spent fuel decay heat is transferred from the fuel assemblies to the tubes in the fuel basket, and through the heat transfer disks to the TSC wall. Heat flows by radiation and convection from the TSC wall to the circulating air and is exhausted through the air outlets. The passive cooling system is designed to maintain acceptable reinforced concrete and peak cladding temperatures for the authorized fuel types during storage.

The top of the VCC is closed by an approximately 5-inch thick shield plug consisting of carbon steel plate (gamma shielding), NS-4-FR (neutron shielding), and a carbon steel lid. The lid is bolted in place and has tamper indicating seals on two of the bolts. The VCC is shown in Figure 1-3.

1.1.3 Transfer Cask

The transfer cask provides shielding during TSC movements between work stations, the VCC, or the NAC-STC transport cask. It is a multi-wall (steel/lead/NS-4-FR/steel) design, about 134 inches in height, with about an 87-inch outer diameter, and weighs about 81,000 lbs empty. The transfer cask has a bolted top retaining ring to prevent a loaded canister from being inadvertently removed through the top of the transfer cask. Retractable (hydraulically operated) bottom shield doors on the transfer cask are used during unloading operations. Clean water is circulated in the gap between the transfer cask and the TSC during spent fuel pool loading operations to minimize contamination of the transfer cask and TSC. The transfer cask is shown in Figure 1-4.

1.1.4 Auxiliary Equipment

Section 1.2.1.4 of the SAR describes the following principal auxiliary equipment necessary to operate the NAC-MPC system in accordance with its design:

- Adapter Plate - mates the transfer cask to the VCC or the NAC-STC transport cask.
- Air Pad Rig Set - allows movement of the VCC on the storage pad.
- Automatic Welding System - minimizes radiation exposure during TSC closure welding.
- Draining and Drying System - used to remove moisture and establish a TSC vacuum.

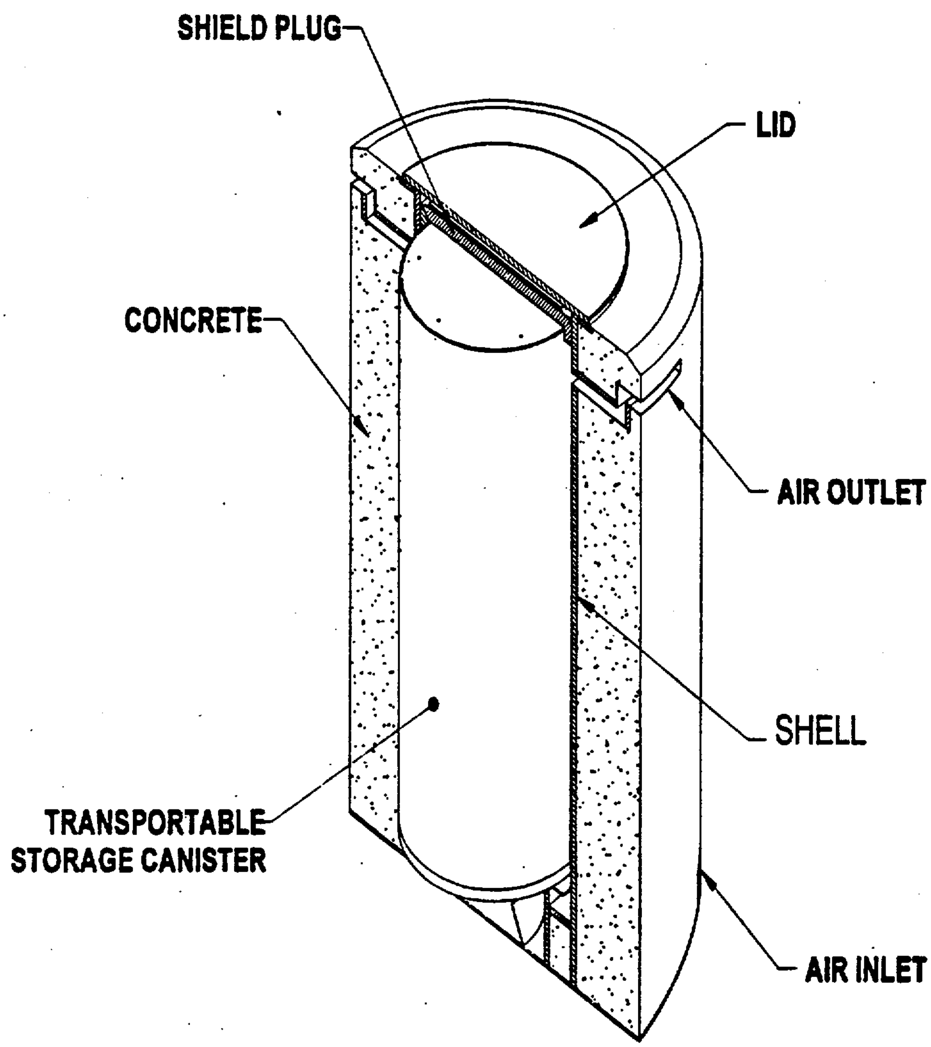

Figure 1-3 Vertical Concrete Storage Cask

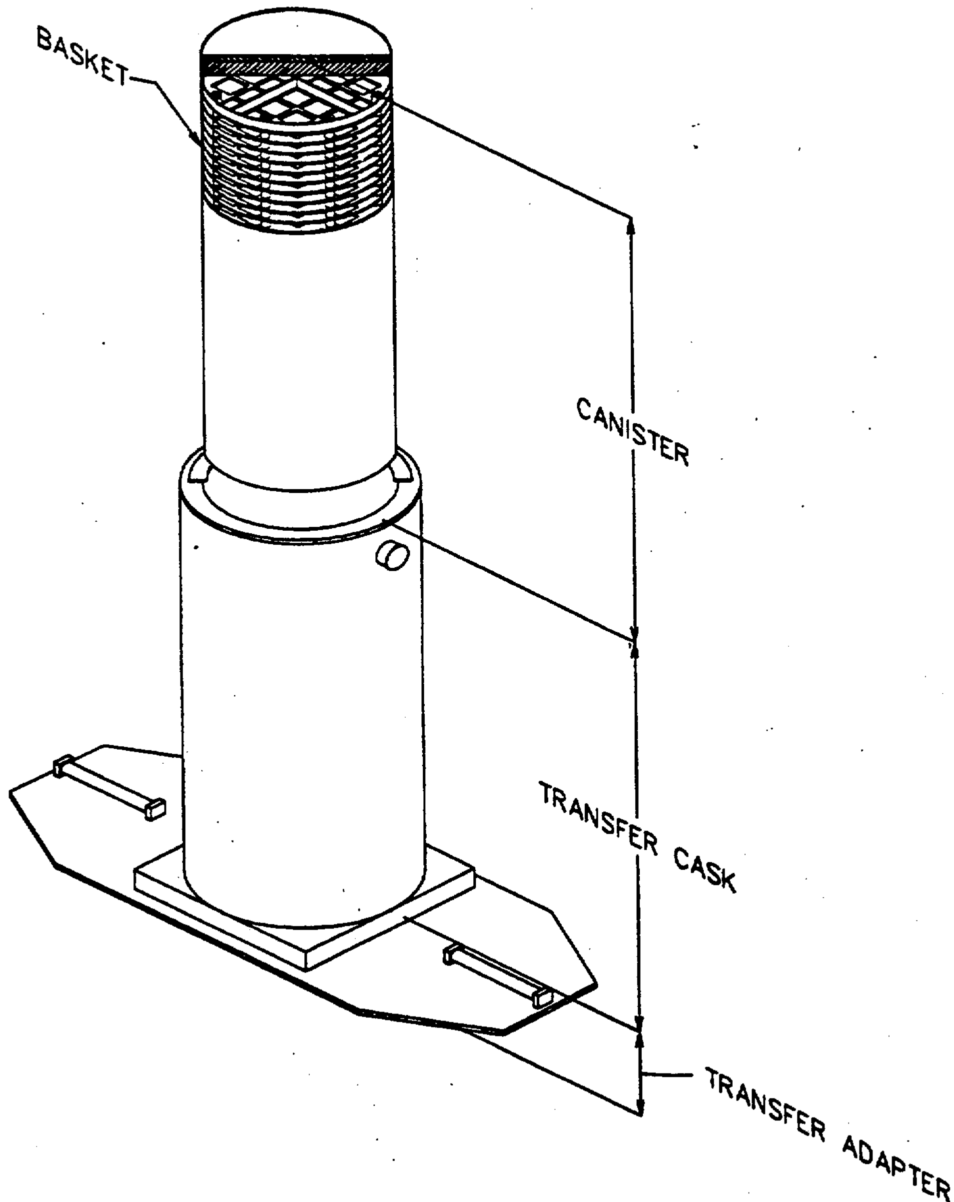

Figure 1-4 Transfer Cask and Canister

- Helium Leak Test Equipment - mass spectrometer to verify the integrity of the shield lid weld.
- Heavy Haul Trailer - used to move the VCC.
- Lifting Jacks - used to lift the VCC to insert/remove the Air Pad Rig Set.
- Riggings and Slings - provided for major components such as shield and structural lids and the transfer cask.
- Temperature Instrumentation - located at VCC outlets and one VCC inlet for local and/or remote temperature indications.

1.1.5 NAC-MPC Storage Cask Arrays

Section 1.4 of the SAR describes and depicts a typical storage pad layout for an Independent Spent Fuel Storage Installation (ISFSI). Spacing limitations on cask arrays (15 feet minimum) are specified in Design Specification 4.5.1 of the Technical Specifications (TS). TS 3.2.1 controls the maximum allowable surface dose rates for any individual cask.

1.2 Drawings

The drawings associated with the NAC-MPC structures, systems, and components (SSCs) important to safety are provided in Section 1.5 of the SAR. Sufficiently detailed drawings regarding dimensions, materials, and specifications were provided by the applicant and allow a thorough evaluation of the entire system. Specific SSCs are evaluated in Sections 3 through 14 of this SER.

1.3 Cask Contents

The approved contents for the NAC-MPC are specified in the TS. The NAC-MPC is designed to store up to 36 intact Yankee Class (PWR) spent fuel assemblies, which are manufactured by Westinghouse, United Nuclear, Exxon, and Combustion Engineering. The assemblies vary in initial enrichment from 3.5 to 4.94 wt% ^{235}U. Unenriched fuel assemblies were not evaluated by the applicant and are not allowed as approved contents.

A TSC may also contain one or more Reconfigured Fuel Assemblies (RFAs), which are designed and approved to hold Yankee Class spent fuel rods, as intact or damaged fuel or fuel debris. An RFA can accept up to an equivalent of 64 full length spent fuel rods (debris, rod segments or whole rods), held in individual stainless steel tubes in an 8 by 8 array. The array of tubes is positioned in a stainless steel container having the same external dimensions of a standard fuel assembly. The spent fuel is held in the individual tubes, with the tubes supported by a basket assembly within the RFA shell. Both the individual fuel tubes and the RFA are designed to allow draining, drying, and inerting.

The enrichment and physical, thermal, and radiological characteristics of the approved contents are given in the TS. The TS also provide definitions for intact and damaged fuel assemblies and fuel rods, fuel debris, and RFAs.

1.4 Qualifications of the Applicant

NAC is the prime contractor for the NAC-MPC design, and all design and specification activities are performed by NAC. Fabrication of steel and concrete components are specified to be performed by qualified vendors and in accordance with quality assurance (QA) programs meeting the requirements of 10 CFR Parts 71 and 72. Section 1.3 of the SAR adequately details NAC's technical qualifications and previous experience in the area of dry cask storage licensing.

1.5 Quality Assurance

The QA program is evaluated in Section 13 of this SER.

1.6 Evaluation Findings

F1.1 A general description and discussion of the NAC-MPC system is presented in Section 1 of the SAR, with special attention to design and operating characteristics, unusual or novel design features, and principal safety considerations.

F1.2 Drawings for structures, systems, and components (SSCs) important to safety are presented in Section 1.5 of the SAR. Specific SSCs are evaluated in Sections 3 through 14 of this SER.

F1.3 Specifications for the spent fuel to be stored in the dry cask storage system (DCSS) are provided in SAR Section 2.1. Additional details concerning these specifications are presented in Section 2 of the SAR and in the TS which are appended to the CoC.

F1.4 The technical qualifications of the applicant to engage in the proposed activities are identified in Section 1.3 of the SAR and are acceptable to the Nuclear Regulatory Commission (NRC) staff.

F1.5 The QA program is described in Section 13 of the SAR and is evaluated in Section 13 of this SER.

F1.6 The TSC was not reviewed in this SER for use as a transportation cask component. The TSC is designed to be compatible for transport with the currently licensed NAC-STC design and is, therefore, subject to a separate 10 CFR Part 71 certification action. Copies of the SAR and CoC issued under 10 CFR Part 71 are on file with the NRC under Docket No. 71-9235.

F1.7 The staff concludes that the information presented in this Section of the SAR satisfies the requirements for the general description under 10 CFR Part 72. This finding is based on a review that considered the regulation itself, Regulatory Guide 3.61, and accepted dry cask storage practices detailed in NUREG-1536.

2.0 PRINCIPAL DESIGN CRITERIA

The objective of evaluating the principal design criteria related to SSCs important to safety is to ensure that they comply with the relevant general criteria established in 10 CFR Part 72.

2.1 Structures, Systems and Components Important to Safety

NAC presented a summary of the principal NAC-MPC system design criteria in Table 2-1 of the SAR. Each NAC-MPC storage system component is assigned, in Table 2.3-1 of the SAR, a safety classification based on the components function and an assessment of the consequences of component failure. The component safety classifications are based on the guidance of NUREG/CR-6407, "Classification of Transportation Packaging and Dry Spent Fuel Storage System Components According to Importance to Safety."

2.2 Design Bases for Structures, Systems and Components Important to Safety

NAC's design bases summary for the NAC-MPC system identified the range of spent fuel configurations and characteristics, the enveloping conditions of use, and the bounding site characteristics.

2.2.1 Spent Fuel Specifications

The NAC-MPC is designed to store up to 36 intact Yankee Class spent fuel assemblies. Table 2.1-1 of the SAR and the TS provide detailed fuel assembly characteristics for the authorized contents. These characteristics include: manufacturer, assembly array, physical assembly dimensions, maximum and minimum enrichments, maximum burnup, minimum cool time, maximum decay heat, total weights per assembly, and initial uranium weight per assembly. Detailed parameters regarding the configuration of individual fuel rods, fill gas pressures, and fuel assembly hardware are also provided.

The NAC-MPC is also designed to store up to 36 RFAs containing intact or damaged fuel and/or fuel debris. The physical parameters of the RFA, including maximum uranium content and enrichment, fuel rod and encapsulating rod dimensions, and the maximum rods per assembly are provided in Table 2.1-2 of the SAR. Canister positions not containing an RFA can be used to store intact Zircaloy, intact stainless steel clad, or a mixture of intact Zircaloy and stainless steel clad fuels as limited by the maximum fuel content weight limit. The RFA contents, mixed RFA and intact clad contents, and maximum contents weight limits are specified in the TS.

Fuel assemblies shall not contain empty fuel rod positions. Any fuel rod that has been removed shall be replaced by a solid Zircaloy or stainless steel rod, as specified in the TS. Furthermore, control components are not authorized for storage, as specified in the TS.

Section 2.1.1 of the SAR specifies the bounding fuel types for the criticality and shielding evaluations and provides the design bases maximum decay heat load. The enrichments, burnups, decay heat rates, and cooling times vary for different fuel types, based on the bounding shielding and thermal evaluations, and are specified in the TS. Sections 3 through 14 of this SER evaluate the bounding fuel types.

2.2.2 External Conditions

Section 2.2 of the SAR identifies the site environmental conditions and natural phenomena for which the storage system is analyzed during the period of storage. The SAR presents analyses that demonstrates that the NAC-MPC system meets the design criteria in subsequent SAR sections, and which is further evaluated in Sections 3 through 14 of this SER.

SAR Sections 2 and 11 identify the normal, off-normal, and accident conditions for which the NAC-MPC design has been evaluated. The staff's evaluation of the system's response to off-normal and accident conditions is located in Section 11 of this SER. The TS, in Section 4.4, identifies the site-specific parameters and analyses that are required to be verified by the NAC-MPC system users.

2.3 Design Criteria for Safety Protection Systems

A summary of the Safety Protection Systems is provided in Section 2.3 of the SAR. These systems are further evaluated in Sections 3 through 14 of this SER.

2.3.1 General

The design life of the NAC-MPC system is 50 years. The codes and standards of design and construction of the system are specified in Section 3.1.2 of the SAR. The transfer cask lifting yoke is single-failure proof and designed to meet the requirements of American National Standards Institute (ANSI) N14.6 and NUREG-0612. The lifting yoke is (1) proof load tested to 300 % of design load upon fabrication and (2) visually inspected prior to each use and inspected annually.

2.3.2 Structural

The structural design criteria are presented in SAR Section 2.2, with combined loadings addressed in Section 2.2.5 of the SAR. The design basis earthquake (DBE) is defined, at the top of a storage pad, as 0.25 g horizontal and 0.167 g vertical seismic accelerations. Confinement of the stored radioactive material is assured by adequate margins of safety during normal, off-normal, and accident conditions and is evaluated in Sections 3.4.4, 11.1, and 11.2 of the SAR, respectively. Similarly, adequate structural margins of safety ensure adequate criticality control as evaluated in Section 6 of the SAR. The thermal analysis (Section 4 of the SAR) determined that cladding temperatures will not exceed the specified maximum allowable cladding temperatures, thereby, protecting the fuel cladding against degradation during storage.

2.3.3 Thermal

The passive heat removal capabilities of the NAC-MPC are described in Section 4 of the SAR. Operating limits and verification are established in the TS to ensure continued safe operation. A remote temperature monitoring system is used to measure the outlet air temperature of the system during long-term storage. The outlet temperature is recorded daily to check the thermal performance of the cask. The VCC inlets and outlets VCC are inspected daily for blockages.

2.3.4 Shielding/Confinement/Radiation Protection

The NAC-MPC confinement system is closed by welding. The TSC shield lid weld is pressure tested, liquid penetrant examined, and leak tested to 8.0 x 10^{-8} cubic centimeters per second (helium). The closure weld of the TSC structural lid is redundant to the shield lid and port cover closures and is also liquid penetrant examined. The longitudinal, girth, and bottom welds of the TSC shell are full penetration welds that are radiographed or ultrasonically inspected during fabrication. None of the evaluated normal, off-normal, or accident conditions result in a breach of the TSC. The TSC is designed to withstand hypothetical accident condition drops in a transportation cask without precluding ready retrieval of the spent fuel.

Personnel radiation exposure during handling and closure of the TSC is minimized by both design features and operational procedures. The shield lid is placed on the TSC while the transfer cask is under water in the spent fuel pool. Clean water is injected into the TSC/transfer cask annulus to minimize contamination of the transfer cask and TSC while submerged in the spent fuel pool. The exterior of the transfer cask is decontaminated prior to draining the TSC to preserve the shielding benefit. The retaining ring on the transfer cask prevents inadvertent lifting of the TSC beyond the transfer casks shielding. Access to the ISFSI is controlled in accordance with the requirements of 10 CFR Parts 20 and 72. The shielding associated with the system design meets the requirements of 10 CFR 72.104 and 72.106 for normal and accident conditions, respectively.

2.3.5 Criticality

Neutron poison sheets (Boral) are attached to each side of each fuel tube in the basket design. The poison sheets and basket design ensure that the cask remains subcritical under normal, off-normal, and accident conditions.

2.3.6 Operating Procedures

The operating procedures descriptions are discussed in Section 8 of the SAR and include procedures for wet and dry loading and unloading operations. Radiation protection design features, including features to facilitate decontamination, are incorporated in both the physical design and the operating procedures.

2.3.7 Acceptance Tests and Maintenance

The acceptance tests and maintenance of the NAC-MPC system are described in Section 9 of the SAR, including the commitments, industry standard, and regulatory requirements used to establish the acceptance, maintenance, and periodic surveillance tests.

2.3.8 Decommissioning

Decommissioning of the NAC-MPC system is described in Section 2.4 of the SAR and evaluated in Section 14 of this SER.

2.4 Evaluation Findings

F2.1 The staff concludes that the principal design criteria for the NAC-MPC are acceptable with regard to demonstrating compliance with the regulatory requirements of 10 CFR Part 72. This finding is based on a review that considered the regulation itself, appropriate regulatory guides, applicable codes and standards, and accepted engineering practices. More detailed evaluations of design criteria and assessments of compliance with those criteria are presented in Sections 3 through 14 of this SER.

3.0 STRUCTURAL EVALUATION

This section evaluates structural performance of the NAC-MPC system for compliance with 10 CFR Part 72. Structural design features, including system weight and center of gravity, mechanical properties of materials, and cask general standards are reviewed. Also evaluated are design basis loads and load combinations, design criteria codes and standards, and analysis results against stress allowables and structural failure modes under the normal, off-normal, and accident conditions and the natural phenomena events.

3.1 Structural Design

3.1.1 Structural Design Features

Section 1 of this SER provides a general description of the NAC-MPC system consisting of three principal components: (1) the TSC (canister), (2) the transfer cask, and (3) the VCC (cask). The major structural design features of these components are as follows.

3.1.1.1 Transportable Storage Canister

The TSC assembly features a circular cylindrical shell with a welded bottom plate, a fuel basket, a shield lid, two penetration port covers, and a structural lid.

The canister shell is fabricated from 5/8-in-thick Type 304L stainless steel plate, joined at its edges by a full penetration weld, which is radiographed. The bottom closure is a 1-inch-thick Type 304L stainless steel plate joined to the canister shell by a full penetration weld, which is ultrasonically examined. The canister shell plus the bottom plate, port covers, and lids constitute the confinement boundaries.

The fuel basket is an assembly of 36 fuel tubes laterally supported by 22 stainless steel disks which are axially retained by the spacers aligned on 8 radially located tie rods. A top weldment and a bottom weldment are attached to the basket ends to support and position the fuel tubes. Type 304 stainless steel is used to construct the fuel tubes, which, together with the encased Boral sheets on all tube outside surfaces, provide spent fuel criticality control in the basket.

Aluminum heat transfer disks, spaced midway between the support disks, are provided for the fuel basket to facilitate heat conduction from the fuel assembly to the TSC wall. Holes in the heat transfer disks are sized to allow fuel tubes and tie rods to pass through to prevent the disks from becoming load bearing members for any structural loads other than dead loads.

The stainless steel RFA provides a structural restraint to maintain the configuration of damaged fuel in its analyzed envelope. It consists of a square tube with end fittings, a basket assembly, and 64, ½-inch-diameter, fuel tubes. To allow the RFA to be handled in the same manner as a standard spent fuel assembly, the external dimensions of the RFA shell are the same as those of a standard spent fuel assembly.

3.1.1.2 Transfer Cask

The transfer cask is a multi-wall circular cylindrical construction for loading the TSC into the VCC. It is equipped with a set of retractable shield doors at the cask bottom and a pair of lifting trunnions at the top. The transfer cask also incorporates a bolted-in-place retaining ring to prevent a loaded TSC from being inadvertently removed out of the top of the transfer cask.

3.1.1.3 Vertical Concrete Cask

The VCC is a storage overpack of reinforced concrete wall construction with a heavy structural steel inner liner. It is closed at the top by a shield plug. In addition to providing structural support to the TSC, the concrete wall serves as a barrier which provides protection for the TSC and its contents in natural phenomenon events, such as tornado wind and tornado generated missiles.

3.1.2 Structural Design Criteria

SAR Sections 2.2 and 3.1.2 summarize the NAC-MPC structural design criteria. The criteria address, in general, the applicable codes and standards, individual loads as related to environmental conditions and natural phenomenon events, load combinations, and stress allowables for normal, off-normal, and accident-level conditions. As evaluated below, the structural design criteria are consistent with those of NUREG-1536 and are acceptable.

3.1.2.1 Codes and Standards

The TSC, as a confinement boundary, is designed per American Society of Mechanical Engineers (ASME) Code, Section III, Subsection NB. The fuel basket component stresses are evaluated in accordance with ASME Code, Section III, Subsection NG and for buckling with NUREG/CR-6322. ANSI N14.6 and NUREG-0612 are used for evaluating the transfer cask lifting trunnions. ANSI/American Nuclear Society (ANS) 57.9 or equivalent is considered for evaluating other transfer cask components, including the retaining ring and its fastening bolts. The RFA is evaluated in accordance with ASME Code, Section III, Subsections NF and NG, and Appendix F for stresses and NUREG/CR-6322, for buckling. The reinforced concrete of the VCC is designed and constructed to the respective American Concrete Institute (ACI) 349 and 318 requirements.

The use of the codes and standards for the NAC-MPC system is consistent with the guidance in NUREG-1536 and is acceptable.

3.1.2.2 Site Environmental and Natural Phenomenon Loads

The SAR considers the pressure, temperature, and mechanical loads typically associated with operating the NAC-MPC system under normal and off-normal conditions for the components structural evaluation. In the following text, the staff reviews the bases for the environmental and natural phenomenon loads, which can be considered as general license site parameters.

Tornado Wind and Tornado Driven Missiles. SAR Table 2.2-1 presents the tornado wind characteristics. The design basis tornado wind loadings are in accordance with the Regulatory Guide 1.76, Region I, tornado with a maximum rotational wind velocity of 290 mph and translational velocity of 70 mph for a maximum combined velocity of 360 mph.

SAR Section 2.2.1.3 lists three types of tornado generated missiles that could impact the cask at normal incidence. In NUREG-0800, Section 3.5.1.4, Spectrum I, the three design basis missiles are: (1) a massive deformable missile of 3960 lbs, (2) a penetration missile of 275 lbs, and (3) a protective barrier missile of a 1-inch diameter solid steel sphere. All missiles are assumed to impact the VCC horizontally at a velocity of 126 mph per hour, which is 35 % of the maximum combined velocity of 360 mph. For missile impact in the vertical direction, the assumed missile velocity is 88.2 miles per hour, which is 70 % of the velocity of a horizontal missile. The missile velocities are consistent with NUREG-0800, Section 3.5.1.4, guidance.

Flood. SAR Section 2.2.2.3 considers a maximum allowable flood water velocity and a maximum allowable flood water depth for evaluating the NAC-MPC system. For a flood water depth of 50 feet above the base of the VCC, SAR Section 11.2.6 calculates a hydrostatic pressure of 22 psig to be exerted on the TSC and VCC. At a water velocity of 15 feet per second, the SAR calculates a drag force of 21,720 lbs for the VCC stability analysis against sliding and tipover. The hydrostatic pressure and drag force effects, as presented in SAR Section 11.2.6, are reviewed in SER Section 3.4.1.

Earthquake. SAR Section 2.2.3.1 states that the maximum seismic accelerations to which the NAC-MPC may be subjected are site specific. The peak acceleration of each of the two horizontal components of the earthquake motion is assumed to be 0.25 g, and the vertical component is assumed to be 0.167 g, all defined as the motion at the top of a concrete storage pad for the SAR Section 11.2.2 seismic stability evaluation against cask tipover and sliding.

Snow and Ice. SAR Section 2.2.4 considers the ANSI/ASCE 7 snow load criteria. On the basis of the exposure, thermal, and importance factors, a design snow and ice load of 100.8 psf is determined for the VCC. The staff agrees with the SAR conclusion that the snow load effect is bounded by the case of applying the weight of the loaded transfer cask to the top of the VCC. As a result, no additional staff evaluation is necessary for snow and ice load effects.

3.1.2.3 Load Combinations

SAR Section 2.2.5 describes the load cases for evaluating the combined load effects on the structural performance of the NAC-MPC system. SAR Table 2.2-3 lists the load combinations for the TSC. In addition to the environmental conditions and natural phenomenon events, the loads considered include the dead weight, live load, thermal effects, internal pressure, handling load, and cask drop and tipover accident loads. SAR Table 2.2-2 summarizes the load combinations for the VCC designed by the factored load method per ACI 349, which is consistent with ANSI/ANS 57.9.

3.1.2.4 Stress Allowables

SAR Table 2.2-4 lists structural evaluation criteria for the TSC components. The stress allowables are based on ASME Code, Section III, Subsections NB and NG and Appendix F.

The stress design factors for the lifting devices are in accordance with ANSI N14.6 and NUREG-0612. The basket component buckling criteria are per NUREG/CR-6322.

SAR Section 2.2.5.2 considers concrete strength reduction factors, in accordance with ACI 349, for the VCC evaluation. SAR Section 2.2.5.4 considers ANSI N14.6 and NUREG-0612 for the lift trunnions and ANSI/ANS 57.9 for the balance of the transfer cask. The stresses in the trunnions are evaluated for six times and ten times the fully loaded weight of the transfer cask for the respective yielding and ultimate material strengths.

3.1.3 Weights and Centers of Gravity

SAR Table 3.2.1 lists the calculated weights and centers of gravity of the major components and total system of the NAC-MPC. The center of gravity locations are identified along the cask vertical axis. The weight of the loaded transfer cask at 143,013 lbs provides the basis for evaluating transfer cask lifting. The total weight of the loaded VCC at 206,094 lbs provides the basis for evaluating structural stability against cask sliding and overturning in the accident-level conditions.

3.1.4 Materials

The applicant provided a general description of the materials of construction in SAR Sections 1.2 and 3.1. Additional information regarding the materials, fabrication details, and testing programs can be found in SAR Sections 7.1, 9.1, and 12.1. The staff reviewed the information contained in these sections and the information presented in the drawings to determine whether the NAC-MPC cask system meets the requirements of 10 CFR 72.24(c)(3) and (4); 72.122(a), (b), (c), (h), and (l); and 72.236(g) and (h). In particular, the following aspects were reviewed: materials selection; applicable codes and standards; weld design and specification; bolt fabrication and preparation; chemical and galvanic reactions; coatings and, long-term cask performance issues, such as delayed cracking, brittle failure, cladding creep, corrosion, lead slumping, changes in toughness, and thermal aging.

3.1.4.1 Structural Materials

Most of the structural components of the TSC (e.g., shell, bottom plate, shield lid, structural lid, basket fuel tubes, etc.) are fabricated from Types 304 or 304L austenitic stainless steel. These types of steels were selected because of their high strength, ductility, resistance to corrosion and metallurgical stability. Because there is no ductile-to-brittle transition temperature in the range of temperatures expected to be encountered for these steels, their susceptibility to brittle fracture is negligible. The TSC basket support disks are fabricated from Type 630 (H1150) precipitate-hardened steel. This type of steel is heat treated to produce higher strengths (e.g., yield and ultimate tensile strengths) than those of Type 304 steel without a significant loss of corrosion resistance. The ductile-to-brittle transition temperature of Type 630 steel is below the expected operating temperatures, so brittle fracture of this material is not expected. Table 1.2-2 of the SAR provides a summary of the fabrication specifications for the TSC including welding, packaging, and QA requirements. The staff concludes that the selection of these materials is acceptable for use in the TSC.

The main structural components of the VCC are fabricated with reinforced concrete and carbon steel. The VCC liner, lid, shield plug, and base weldment are fabricated from American Standard Testing and Materials (ASTM) A 36 steel, a commonly used steel for structural applications. The 21-inch thick, reinforced concrete shell has a minimum specified compressive strength and density of 4000 psi and 140 lb/ft^3, respectively. The cement used to fabricate the concrete shell is Type II Portland cement meeting the requirements of ASTM C150; the reinforcing steel is an ASTM A615, Grade 60 steel; and the concrete aggregate meets the specifications of ASTM C33. Other fabrication specifications for the VCC, including the QA specifications, can be found in Table 1.2-4 of the SAR. The staff concludes that the concrete materials meet the requirements of ACI 349, and the materials comprising the VCC are suitable for structural support, shielding, and protection of the TSC from environmental conditions.

Transfer cask structural components (including the inner and outer shells, trunnions, shield doors, etc.) are primarily fabricated with either ASTM A 588 or ASME SA 350 high-strength, low-alloy steel. These types of steels are common structural materials and have ductile-to-brittle transition temperatures below the TS minimum temperature operating limit of 0 degrees Fahrenheit (°F). The staff concludes that these steels are suitable for use in the transfer cask.

3.1.4.2 Nonstructural Materials

Criticality control in the TSC basket is achieved by surrounding each fuel assembly with stainless steel clad Boral sheets. Boral has a long, proven history in worldwide nuclear service and has been used in other spent fuel storage and transportation casks. The Boral sheets are completely enclosed within the welded stainless steel cladding to prevent its exposure to environments that could otherwise cause it to degrade. In accordance with Section 9.1.6 of the SAR, wet chemistry and/or neutron attenuation techniques will be used to ensure that the Boral sheets have a minimum ^{10}B loading of 0.01 grams/cm^2 ^{10}B.

Neutron absorbers and gamma shields should be fabricated from materials that can perform well under all conditions of service during the license period. Interlocking chemical lead bricks (ASTM B29) are used in the transfer cask for gamma shielding. The thermal analyses of SAR Section 4 and SAR Table 4.1-4 show that the temperatures of the lead bricks during transfer operations (e.g., 191°F) are well below the melting point of this material (e.g., 600°F). Therefore, the staff concludes that the lead will undergo minimal slumping and will perform its intended function of gamma shielding. The transfer cask also utilizes NS-4-FR neutron shielding. The NS-4-FR material is a high-hydrogen content, durable, fire resistant material that has been used reliably in several other storage cask systems. As SAR Table 4.1-4 shows, the temperatures experienced by the NS-4-FR are about 100°F lower than the temperature limit. As noted above, concrete provides neutron and gamma shielding during storage. The thermal analyses of SAR Section 4 indicates that the peak temperature of the concrete will be well below the ACI 349 prescribed temperature limits. The staff concludes that the chemical lead bricks, the NS-4-FR neutron shielding material, and the concrete in the VCC are suitable shielding materials for the NAC-MPC system.

3.1.4.3 Welds

The TSC shell is assembled using full penetration longitudinal and girth welded joints in the shell and circumferential welded joints at the junction between the bottom plate and the shell.

These welds are performed in accordance with ASME Code Section III, Subsection NB-4000. The shield and structural lids are joined to the shell by partial penetration welds. The applicant has taken an exception to ASME Code, Section III with respect to the design of this redundant closure. Visual (VT), radiographic (RT), ultrasonic (UT), and liquid penetrant (PT) examination requirements of these welds are summarized in Section 9 of this SER. All welding of components of the basket assembly are performed in accordance with ASME Code, Section III, NG-4000. Welding of other NAC-MPC components (e.g., transfer cask and VCC) will be performed in accordance with either American Welding Society (AWS) D1.1-96 or ASME Code, Section VIII. All exceptions to the ASME Code are identified in the TS.

The NAC-MPC materials of construction (e.g., stainless, carbon, low alloy steels, etc.) are readily weldable using commonly available welding techniques. The cask welds were well-characterized on the drawings, and standard welding symbols and notations in accordance with AWS Standard A2.4, "Standard Symbols for Welding, Brazing, and Nondestructive Examination", were used.

The staff concludes that the welded joints of the TSC, transfer cask, and VCC meet the requirements of the ASME and AWS Codes, as applicable. Although the TSC closure welds are partial penetration welds, this configuration will perform its intended structural and confinement functions.

3.1.4.4 Bolting Materials

The TSC is an all-welded canister. The only bolts used for structural applications of the NAC-MPC cask system are the retaining ring bolts on the transfer cask. They are fabricated from ASME SA-193 ferritic high-strength steel. The ductile-to-brittle transition temperature of this steel is below the expected operating temperatures, so brittle fracture of the bolts is not expected. Procurement of the bolts in accordance with the ASME SA-193 specification will help to ensure that the material receives the proper heat treatment and possesses the required mechanical properties. The staff finds the bolting material acceptable.

3.1.4.5 Coatings

No zinc, zinc compounds, or zinc-based coatings are used in the NAC-MPC system. All of the exposed surfaces of the transfer cask and VCC will be coated with an acceptable epoxy enamel coating which has been widely used in the nuclear industry. This coating will protect the steel from excessive oxidation and facilitate decontamination of the surfaces.

3.1.4.6 Mechanical Properties

Tables 3.3-1 through 3.3-13 of the SAR provide mechanical property data for the major structural materials including stainless steels, precipitation-hardened steel, carbon steel, bolting materials, aluminum alloys (even though the aluminum components are not considered to be structural members), concrete, and NS-4-FR neutron shielding material. Most of the values in these tables were obtained from ASME Code, Section II, Part D. However, some of the values were obtained from other acceptable references. The staff independently verified the temperature dependent values for the stress allowables, modulus of elasticity, Poisson's ratio, weight density, and coefficient of thermal expansion. The staff concludes that these material

properties are acceptable and appropriate for the expected load conditions (e.g., hot or cold temperature, wet or dry conditions) during the license period.

3.1.5 General Standards for Cask

SAR Section 3.4 performs structural analyses of the NAC-MPC components, under normal operating and certain off-normal temperature conditions, to meet the general standards for storage casks. The analyses also address the chemical and galvanic reactions and positive closure of the system. SAR Sections 11.1 and 11.2 consider structural performance of the NAC-MPC under off-normal events and accident-level conditions. The analyses are used to demonstrate the structural capabilities that are relied on to preclude (1) unacceptable risk of criticality, (2) unacceptable release of radioactive materials to the environment, (3) unacceptable radiation dose to the public or workers, and (4) significant impairment of the ready retrievability of stored nuclear materials. The SAR analyses are evaluated in SER Sections 3.2, 3.3, 3.4, and 11.

3.1.6 Supplemental Data

3.1.6.1 Finite Element Analysis Codes

The SAR uses three general purpose finite element codes, ANSYS, STARDYNE, and LS-DYNA, to perform structural analysis of the NAC-MPC system. The stresses in the canister, fuel basket, transfer cask trunnion-to-shell interface, and concrete overpack are analyzed with ANSYS, a widely used code in the nuclear power industry. STARDYNE is only used to analyze the RFA basket tie plate. The VCC tipover decelerations are calculated with LS-DYNA, which is a PC-based version of the DYNA3D code considered in NUREG/CR-6608. All three codes are commercially available.

3.1.6.2 Finite Element Structural Analysis Models

SAR Sections 3.4.3, 3.4.4, 11.2.12, 11.3.1, and 11.3.2 provide details of the ANSYS finite element models of the NAC-MPC components subject to various pressure, temperature, and mechanical loading conditions. For ANSYS element selection, the SAR uses SOLID45 to model the canister shell and closure plates, SHELL63 for the basket support disks and weldments, SHELL93 and SOLID95 for the transfer cask trunnion-to-shell joint, PLANE42 for the axisymmetric concrete overpack, and CONTAC52 and COMBIN40 for the gap opening and closing between individual structural entities. Temperature-dependent material properties are considered in finite element modeling. As loading and component configurations permit, appropriate quarter- or half-symmetry model configurations are used. Additionally, for canister lifting and tipover accident stress analyses, only the top portions of the TSC are used for the structural behavior simulation. The SAR defines design margin or margin of safety as the ratio of the stress allowable and the calculated stress minus one, for which the at-temperature stress allowables are considered.

The staff reviewed the SAR finite element modeling approaches, including finite element scheme, element selection, loading application, and boundary conditions, and concludes that they are acceptable for the finite element structural analysis of NAC-MPC components. The implementation of these approaches and additional analysis assumptions, as appropriate, are also reviewed together with the SAR analysis results in the following sections.

3.2 Normal Operating and Design Conditions

3.2.1 Chemical and Galvanic Reactions

In Section 3.4 of the SAR, the applicant evaluated whether chemical, galvanic, or other reactions among the materials and environments would occur. The staff reviewed the design drawings and applicable sections of the SAR to evaluate the effects, if any, of intimate contact between various NAC-MPC system materials of construction during all phases of operation. In particular, the staff evaluated whether these contacts could initiate a significant chemical or galvanic reaction that could result in component corrosion or combustible gas generation. Pursuant to NRC Bulletin 96-04, a review of the NAC-MPC system, its contents, and operating environments has been performed to confirm that no operation (e.g., short-term loading/ unloading or long-term storage) will produce adverse chemical or galvanic reactions.

The TSC is primarily fabricated with stainless steel and aluminum. During storage, it will be backfilled with helium (99.9% minimum purity) cover gas. Using this level of helium purity, a maximum of 0.3 gram-moles of oxidizing gases could exist in the cask for all loading conditions. This amount is well below the 1.0 gram-mole limit that is recommended in PNL-6365 and specified in NUREG-1536. The vacuum drying procedures of SAR Section 8.1.1, i.e., two cycles of sequentially evacuating and backfilling the TSC with helium, and the careful design, configuration, and operation of the vacuum drying equipment will ensure that contamination of the cover gas with air is minimal. The staff concludes that in this dry, inert environment the TSC components are not expected to react with one another or with the cover gas. Further, corrosion or oxidation of the fuel and TSC internal components will effectively be eliminated during storage.

The applicant identified one potential chemical or galvanic reaction between the aluminum heat transfer disks of the TSC and the relatively low pH (e.g., 4.0-4.5), borated water in PWR spent fuel pools. This reaction may produce small amounts of combustible gases, such as hydrogen, during loading and unloading operations. The safety hazards associated with ignition of this hydrogen gas are mitigated by employing the procedures of SAR Sections 8.1 and 8.3 which specify that the user monitor the concentration of hydrogen gas during any welding or cutting operations. If hydrogen gas is detected at concentrations above 2.4% in air at anytime during these operations, the hydrogen gas will be removed by flushing the suspect regions with ambient air before continuation of the welding process. The staff concludes that these procedures are adequate to prevent ignition of any hydrogen gas that may be generated during welding operations. Further, the potential reaction of the aluminum with the spent fuel pool water will not impact the ability of the aluminum heat transfer disks to perform their intended heat transfer function because the loss of aluminum material is expected to be negligible.

The transfer cask is constructed from carbon and low alloy steels, NS-4-FR polymeric material, and lead. An epoxy enamel coating will be applied to all of the exposed surfaces of the transfer cask to minimize corrosion of its carbon steel components. The staff concludes that none of the transfer cask materials or the coating are expected to degrade, react with each other, or react with TSC components or fuel during the loading and unloading operations.

3.2.2 Positive Closure

The TSC is a structure system with multipass welds to join the canister shield lid and structural lid to the shell. The penetrations to the canister cavity, through the shield lid, are closed by welded port covers. These design features preclude inadvertent opening of the TSC. The top of the VCC is closed by a bolted lid weighing approximately 3,000 lbs. The weight of the lid, its inaccessibility, and the presence of the bolts effectively preclude inadvertent opening of the lid. The staff concurs with the SAR conclusion that the NAC-MPC employs an acceptable positive closure system.

3.2.3 Lifting Devices Analysis

SAR Section 3.4.3 evaluates the lifting devices of the NAC-MPC. The transfer cask is lifted by two trunnions. The loaded and closed TSC is lifted with two three-legged slings, through the six hoist rings threaded into the structural lid. The VCC is raised by four lifting jacks placed at the jacking pads near the end of the overpack air inlets. The structural performance of these lifting devices is evaluated as follows.

3.2.3.1 Transfer Cask Lift

Two 10-inch-diameter trunnions are welded to the transfer cask body at the inner and outer shells. SAR Section 3.4.3.3 states that the trunnion design meets ANSI N14.6 and NUREG-0612 for heavy lifts. For structural analysis, the transfer cask is assumed to weigh 150,000 lbs, which envelops the maximum weight of the loaded transfer cask. SAR Section 3.4.3.3 describes the finite element analysis of the trunnion-to-shell joint of the transfer cask. SAR Tables 3.4.3.3-1 and -2 summarize stress results for the transfer cask outer and inner shells, respectively. Except for localized over-stresses, as permitted by ANSI N14.6, stresses in the shells are shown to meet the stress design factors of 6 and10 against the respective yielding and ultimate strengths. The SAR calculates a maximum trunnion bending stress of 3,287 psi, which corresponds to the stress design factors of 9.7 and 21.3 against the respective material yield and ultimate strengths, and is acceptable.

The SAR evaluates other load bearing components of the transfer cask for the loading conditions and stress allowables commensurate with the postulated design events. This includes the bolted-in-place retaining ring assembly evaluated for inadvertent TSC lifting. The shield door assembly at the bottom of the transfer cask is shown to have stress design factors larger than 6 and 10 against the yield and ultimate strengths, respectively.

The staff concurs with the SAR conclusion that the transfer cask is structurally adequate in meeting the non-redundant, heavy-lifting requirements of ANSI N14.6 and NUREG-0612.

3.2.3.2 Transportable Storage Canister Lift

SAR Section 3.4.3.2 evaluates structural performance of the hoist rings, the structural lid, and the weld that joins the structural lid to the shell for lifting a loaded TSC of 54,730 lbs. Six hoist rings, each at the rated capacity of 24,000 lbs or ultimate capacity of 120,000 lbs, are used with two three-legged slings for a single-failure-proof redundant lift. On the basis of the minimum allowable distance of 67 inches, measured from the master link of the sling to the top of the canister, the SAR determines a load factor of 10.9 for the hoist ring, which is acceptable.

The SAR evaluates the 1-1/2 inch diameter hoist ring bolts, considering strengths of the mating bolt and structural lid materials. The evaluation demonstrates that the bolt engagement length of 2.5 inches is adequate to preclude thread failure in the structural lid bolt holes.

The SAR models the upper portion of the TSC to evaluate the structural lid and its weld to join the TSC shell. The hoist ring force applied to the structural lid is simulated with a three-point lifting configuration, in lieu of the six-point lifting specified, and appropriate modeling consideration is given to the boundary condition associated with the truncated canister model and the anticipated symmetric stress distribution. The SAR calculates a maximum stress intensity of 3,753 psi in the structural lid, which corresponds to the stress design factors of 5.40 and 16.9 against the material yield and ultimate strengths, respectively. The results demonstrate that, consistent with Regulatory Guide 3.61, the canister load bearing members are capable of supporting three times the weight of the loaded TSC without generating a stress, in any part of the canister, in excess of the material yield strength.

3.2.3.3 Storage Cask Bottom Lift

SAR Section 3.4.3.1 evaluates structural performance of the VCC components, considering the loading associated with the VCC bottom lift operation. The evaluation includes concrete bearing stresses at the lifting jack locations, size and spacing of Nelson stud anchors for the cask base, and stresses in the TSC support pedestal. The SAR structural evaluation was performed in accordance with ACI 349 and the American Institute of Steel Construction "Manual of Steel Construction," consistent with NUREG-1536. The staff reviewed the SAR results and agrees with the SAR conclusion that sufficient margins exist to demonstrate structural adequacy for the VCC bottom lift operation.

3.2.4 Hot and Cold Temperature Effects

The thermal evaluation of the NAC-MPC system is reviewed in SER Section 4. This section reviews the application of pressure and thermal loadings, on the basis of hot and cold temperature effects, for the structural analysis of the NAC-MPC components. The stress performance resulting from differential thermal expansions is also reviewed. The cold temperature effects on brittle fracture were evaluated in SER Section 3.1.4

3.2.4.1 Internal Pressures and Temperatures

SAR Section 2.2.6 defines the design basis ambient temperatures of 75°, 100°, -40°, and 125°F for the normal, off-normal severe heat, off-normal severe cold, and accident extreme heat conditions, respectively. SAR Section 4.4.5 calculates a maximum canister internal

pressure of 7.9 psig for the normal condition, which provides the basis for applying a design internal pressure of 11.5 psi for the canister structural analysis. SAR Section 3.4.4.1.1 presents a finite element thermal stress analysis of the canister, considering a temperature distribution that envelops the conditions of the 100°F and -40°F ambient temperatures. The analysis considers bounding temperature gradients between key locations on the canister for temperature distribution calculations and subsequent thermal stress analysis. For the fuel basket, SAR Sections 3.4.4.1.8 and 3.4.4.1.9 consider bounding temperatures at the center and circumference of the support disk and the top and bottom weldments to perform temperature distribution and thermal stress analyses. For the VCC, the concrete surface temperatures, as calculated in SAR Section 4.4.1.1, serve as the boundary condition for the heat transfer solution and subsequent stress analysis by means of an axisymmetric two-dimensional finite element model.

The staff concludes that the analyses using bounding pressure and temperature loadings as input to the finite element stress models follow acceptable engineering practice.

3.2.4.2 Differential Thermal Expansion

The SAR evaluates the effects of differential thermal expansion thermal stresses for the TSC and VCC. SAR Table 3.4.4.1-1 shows a maximum canister thermal stress of 13.35 ksi considering a temperature distribution that envelops all normal and off-normal conditions of storage. The combined stresses for the TSC, as evaluated in SAR Section 3.4.4.1.5, are reviewed in SER Section 3.2.5. SAR Section 3.4.4.2.3 considers thermal stress components in the radial, vertical, and circumferential direction for designing the reinforcing steels in the VCC concrete wall. The analysis adequately demonstrates the acceptability of rebar sizes and spacings for the VCC.

3.2.4.3 Cold Temperature

The temperature gradients which may cause thermal stresses to develop in the NAC-MPC components are essentially the same for all of the design basis steady-state ambient temperatures, including the off-normal severe cold condition of -40°F. The canister design internal pressure of 11.5 psig bounds the pressure of 7.9 psig calculated for an ambient temperature which is higher than the off-normal severe cold temperature. Therefore, the SAR has sufficiently considered the cold-temperature thermal and pressure effects for structural evaluation.

3.2.5 NAC-MPC System Components Structural Analysis

SER Section 3.1.6.2 reviews the finite element modeling for the NAC-MPC components. In the following sections, the staff evaluates the SAR Section 3.4.4 analyses of structural performance of system components, under the individual and combined dead weight, thermal, pressure, and handling loads.

3.2.5.1 Transportable Storage Canister

SAR Table 3.4.4.1-1 summarizes the maximum canister thermal stresses under the normal operating condition. SAR Tables 3.4.4.1-2 and -3 present the respective canister primary

membrane and primary membrane-plus-bending stress summaries for the dead weight load. SAR Tables 3.4.4.1-4 and -5 present stress results for the canister subject to an internal pressure of 11.5 psig. SAR Tables 3.4.4.1-6 and -7 list normal handling stresses on the basis of a simulated three-point TSC lifting configuration.

SAR Tables 3.4.4.1-8 through -10 summarize the maximum stresses in the canister for the respective primary membrane, primary membrane-plus-bending, and primary-plus-secondary stresses at key canister locations. A minimum design margin of 0.22 is reported. On this basis, the staff concurs with the SAR conclusion that the structural performance of the canister is acceptable for normal operating conditions.

3.2.5.2 Fuel Basket Support Disk

SAR Section 3.4.4.1.8 analyzes the fuel basket support disk for the storage and handling conditions, using a quarter-symmetry finite element model. The analysis considers the out-of-plane dead weight and the temperatures at the center and around the outer edge of the disk. SAR Table 3.4.4.1-11 summarizes the combined load effects with a minimum design margin of 2.51, which is acceptable.

3.2.5.3 Fuel Basket Top and Bottom Weldments

SAR Section 3.4.4.1.9 analyzes the top and bottom weldments of the fuel basket for the storage and handling conditions. The analysis considers appropriate dead weight loads and support conditions. For the top weldment, the temperature at its center is assumed to be 400°F and its circumference 380°F. For the bottom weldment, the corresponding temperatures are 150°F and 100°F. SAR Table 3.4.4.1-11 summarizes the combined load effects with a minimum design margin of 0.11, which is acceptable.

3.2.5.4 Fuel Tube

The basket fuel tube, which rests on the bottom weldment and serves to position the spent fuel assembly, is free to expand in the axial direction to preclude the development of thermal stresses. In the canister storage configuration, the fuel tube does not serve any load bearing function other than to support its own weight. As a result, stresses due to dead weight load and thermal expansion are negligibly small.

3.2.5.5 Reconfigured Fuel Assembly

The RFA is a thin shell weldment with end fittings to hold a basket assembly for positioning and supporting 64 half-inch diameter stainless steel fuel tubes. SAR Section 11.4 analyzes the structural performance for the off-normal and accident impact loads. Based on the SAR analysis evaluated in SER Section 3.3.2, the staff has reasonable assurance that the RFA is adequately designed for the normal operating condition.

3.2.5.6 Vertical Concrete Storage Cask

SAR Section 3.4.4.2 evaluates the structural performance of the VCC for normal conditions of storage by considering the dead weight, live, and differential thermal expansion loads. SAR

Table 3.4.4.2-1 provides a stress summary for the load combinations defined in SAR Table 2.2-2. The dead weight and live loads concrete stresses are negligibly small. The maximum compressive thermal stresses for the off-normal severe heat condition are 670 and 227 psi in the cask axial and circumferential directions, respectively. SAR Table 3.4.4.2-2 summarizes the maximum concrete stresses and reinforcing bar forces. The minimum margins of safety are 0.21 and 0.33 for the respective vertical and hoop reinforcing bars. On the basis of these results, the staff concludes that the VCC is structurally acceptable for normal operating conditions.

3.3 Off-Normal Events and Accident Conditions

SAR Section 11.0 presents accident analyses, including structural evaluations, to demonstrate that the NAC-MPC satisfies the requirements of 10 CFR 72.24 and 72.122. This section evaluates the SAR analyses of the structural performance of the TSC and RFA under the bounding conditions. These review findings are also considered in the subsequent evaluation of the NAC-MPC components for the off-normal and accident conditions.

3.3.1 Transportable Storage Canister Bounding Structural Analysis

SAR Section 11.3 summarizes the modeling assumptions and structural analysis results for the TSC subject to an individual side drop impact of 55 g and an end drop impact of 56.1 g in its transport configuration. An internal canister pressure of 20 psig is considered for load combination effects.

For the TSC bottom end impact, the fuel and basket weight, as amplified by an inertia load factor of 56.1 g, is applied to the canister bottom plate considered to be fully supported. For the side impact, an amplified inertia load of 55 g is considered. SAR Figure 11.3.1.1-1 depicts schematically the ANSYS finite element model for the end and side impact stress analyses. The staff reviewed the finite element scheme, loading application, and boundary condition for the analysis model, including the gap and contact interfaces between structural entities. The staff concludes that the analysis approach follows standard engineering practices and is acceptable.

SAR Tables 11.3.1.2-1 and -2 present the primary membrane and primary membrane-plus-bending stress results, respectively, for the canister bottom end impact at 56.1 g, with a minimum stress margin of 3.92. SAR Tables 11.3.1.2-3 and -4 present stress results, for the canister side impact at 55 g, with a minimum stress margin of 0.83. SAR Tables 11.3.1.2-6 thru -9 present stress intensity results for the TSC subject to either an end or a side impact force of 20 g. Since the analytical approach is identical to the one already reviewed, the staff agrees that the analysis results can be considered in evaluating the TSC for the impact loads associated with other off-normal events and accident conditions.

SAR Section 11.3.1.3 evaluates the buckling strength of the TSC subject to an end impact load of 56.1 g. The TSC is assumed to be an unsupported, circular cylinder and evaluated with a commonly used formulation for critical buckling load. This results in an acceptable safety margin of 6.3.

SAR Section 11.3.2.1 evaluates the fuel basket support disk subject to an end impact of 56.1 g. The disk is assumed to be supported at the eight split spacer locations to resist the as-amplified dead weight. On the basis of the at-temperature stress allowables and in accordance with the ASME Code, Subsection NG, the SAR calculates a minimum safety margin of 1.42 for the primary membrane-plus-bending stress.

SAR Section 11.3.2.4 presents a fuel basket bottom weldment finite element analysis for an end impact of 56.1 g. The weldment plate, which is supported by eight ribs and eight tie-rod ends, is analyzed for resisting the as-amplified weight of the fuel tubes. On the basis of the at-temperature stress allowables and in accordance with the ASME Code, Subsection NG, the SAR calculates a minimum safety margin of 0.3 for the primary membrane-plus-bending stress.

SAR Section 11.3.2.3 refers to the SAR for the NAC-STC transportation cask, Docket No. 71-9235, for the basket fuel tube evaluation of an end drop of 56.1 g and a side drop of 55 g. The staff reviewed the evaluation and concurs with the SAR conclusion that the fuel tube geometry will be maintained such that the Boral neutron poison will remain in place under the impact conditions analyzed.

The staff reviewed SAR Section 11.3.2.2 and concurs with its conclusion that the fuel basket tie rods and spacers are structurally adequate for a 56.1 g end impact.

3.3.2 Reconfigured Fuel Assembly Bounding Structural Analysis

SAR Section 11.4 evaluates the RFA for off-normal and accident impact conditions. An off-normal inertia force of 20 g is considered for both the side impact and end impact. An accident inertia force of 55 g is considered for the side impact and 57 g for the end impact. In the following, the staff reviewed the SAR evaluation of the RFA components, including the shell casing weldment, basket assembly, and fuel tube.

Except for the basket assembly tie plate, the SAR demonstrates the structural adequacy of the RFA components by long-hand calculations, including the shell casing buckling strength evaluation per NUREG/CR-6322. The stress analysis for the RFA shell is based on its maximum side deflection as limited by the TSC fuel tube geometry. The analysis results meet the ASME Code, Service Levels B and D, criteria for the respective off-normal and accident conditions.

The RFA basket assembly is a weldment of four 97.4-inch-long corner angles joined transversely by seven equally spaced tie plates. For side impact, the SAR shows acceptable stress performance for the corner angles, based on the maximum deflection as limited by the TSC fuel tube geometry. Per NUREG/CR-6322, the SAR also demonstrates adequate margins against buckling for the basket corner angles subject to end impact loads.

The RFA fuel tube is evaluated as a continuous beam with six equal spans. The calculated stress margins are 3.10 and 3.15, based on the ASME Code, Service Levels B and D, allowables, respectively. For buckling under end impact, the SAR demonstrates a margin of 0.74 and 0.13 based on the Service Levels B and D loading conditions, respectively. This is acceptable.

The staff reviewed SAR Sections 11.4.2.5 through 11.4.2.7 for structural performance of the basket tie plate subject to the RFA end impact and the thermal loading effects. Based on the Service Level B stress allowables, the staff concurs with the SAR conclusion that the thermal stress prevails and the tie plate is structurally adequate.

3.3.3 Canister Off-Normal Handling Load

SAR Section 11.1.2 adds to the normal operating loads an inertia load of 0.5 g in each of the three orthogonal directions to account for the TSC handling load effects. The SAR calculates stresses on the basis of the bounding structural analyses which were evaluated in SER Section 3.3.1. The staff concurs with the SAR conclusion that the canister will maintain positive design margin in an off-normal handling accident.

3.3.4 Severe Environmental Conditions

For thermal stress analysis of the VCC, SAR Section 11.1.4 determines that the severe environmental condition, with an ambient temperature of either -40 or 100°F, is bounded by the accident extreme heat condition. SAR Section 11.1.4.2 references SAR Section 3.4.4 in evaluating thermal stresses for the TSC and fuel basket, which considers the bounding temperature distribution for normal and off-normal (severe) environmental conditions. In SER Section 3.2.4.2, the staff determined that the SAR bounding evaluation for differential thermal expansion effects was acceptable. On this basis, the staff concludes that the NAC-MPC system is structurally adequate to withstand the severe and extreme temperature conditions.

3.3.5 Accident Pressurization

Based on a hypothetical failure of all fuel rods, SAR Section 11.2.1 considers an internal pressure of 55 psig for the TSC stress analysis. SAR Tables 11.2.1-1 and -2 list the TSC primary membrane and primary membrane-plus-bending stresses, respectively. The staff agrees with the SAR results and concludes that the TSC is structurally adequate in resisting the postulated internal pressure of 55 psig.

3.3.6 Explosion

The SAR states that an explosion is an unlikely event because administrative controls will exclude explosive substances in the vicinity of an ISFSI. SAR Section 11.2.3 references the SAR Section 11.2.6 analysis which demonstrates acceptable structural performance of the TSC when subjected to an external static pressure of 22 psig. On this basis, the staff concurs with the SAR conclusion that there is no adverse consequences to the canister as a result of an explosion which exerts an equivalent static pressure of less than 22 psig on the TSC.

3.3.7 Storage Cask 6-Inch Drop

SAR Section 11.2.11 evaluates the VCC for a 6-inch drop onto a concrete storage pad. The accident assumes a vertical drop distance of 6 inches, which bounds the normal cask lift of about 3 inches required to install and remove the inflatable air pads for moving the VCC across the surfaces of the transporter and the ISFSI pad. The SAR estimates impact loads by an energy balance method. For the VCC, the cylindrical portion of the concrete is assumed to be

crushed against an infinitely rigid pad. By equating the total potential energy to the energy dissipated through concrete crushing in the VCC, the SAR calculates a maximum deceleration of 169.6 g, which corresponds to a concrete crush depth of 0.036 inch. The staff reviewed the SAR assumptions and concludes that the results are bounding. Similarly, on the basis of the flow stress in a portion of the pedestal weldment, the SAR calculates a maximum end impact deceleration of 19.95 g for the TSC. Since the deceleration is much smaller than the end impact considered in SAR Section 11.3 and reviewed in SER Section 3.3.1, the staff concurs with the SAR conclusion that the TSC is structurally adequate to withstand a 6-inch VCC drop accident.

3.3.8 Cask Tipover

In the following text, the staff reviewed the SAR evaluation of the TSC impact capability under a postulated cask tipover accident.

3.3.8.1 Determination of Deceleration g-Loads

SAR Section 11.2.12.2 presents a finite element analysis of a cask-pad-soil interaction system to compute deceleration g-loads for the NAC-MPC tipping over onto the concrete storage pad. Computer code LS-DYNA is used for modeling the system and performing transient analysis. To demonstrate the adequacy of the analytical method, the SAR follows the NUREG/CR-6608 model validation approach by comparing the computed responses of a billet-pad-soil test system model to the test data. The staff reviewed the SAR and the NAC December 19, 1998, submittal, which provided further clarification of modeling details, and found adequate correlation between the calculated and tested billet responses in pulse amplitude, shape, and duration. On this basis, the staff concludes that the LS-DYNA analysis approach, as implemented by NAC for the billet tipover, is adequately validated.

The SAR adapts the validated billet-pad-soil model for developing the cask-pad-soil interaction model for the NAC-MPC system. By following standard engineering practice, the SAR develops the cask-pad-soil model in a two-step process: (1) all billet test system modeling parameters, including major interaction features such as sliding surfaces between structural entities, are retained, as appropriate, and applicable soil boundary conditions are considered; and (2) the finite element model of the billet itself is replaced with that of the cask. This results in the cask-pad-soil interaction model consisting of a 30 ft x 30 ft x 3 ft concrete pad, a 45 ft x 45 ft x 6 ft soil subgrade, and a circular hollow concrete cylinder with an inner steel liner to calculate cask deceleration g-loads. Other major modeling assumptions include: (1) the pad size based on the tributary area of a typical 30 ft wide by 120 ft long pad to support 16 VCCs in two rows of 8 casks each, (2) the equivalent weights of the system components to be appropriately lumped on the steel liner portion of the model, (3) the soil subgrade configuration to simulate a non-transmitting boundary, (4) the concrete compressive strength of 4,000 psi for both the cask and pad, and (5) the subgrade stiffness of 250 psi/inch, which is less than the site-parameter soil stiffness of 300 psi/inch also shown to be acceptable, for computing the soil subgrade modulus of elasticity.

The SAR simulates an at-rest, center-of-gravity-over-the-corner, tipover initial condition by applying an angular velocity of 1.516 rad/sec to the entire cask model tipping over onto the concrete pad. SAR Section 11.2.12.2.5 summarizes the calculated decelerations of 31.4 g and

27.5 g for the locations corresponding to the canister structural lid and the top most fuel basket support disk, respectively. In its February 23, 1999, submittal, NAC discussed a sensitivity study in which the entire cask was modeled as a rigid body, a bounding cask configuration, to render a cask deceleration of 29.1 g at the top most fuel support disk location. On the basis of previous tipover analysis review experience, the staff evaluated the SAR analysis and the sensitivity study and noted that, as expected, the bounding deceleration of 29.1 g is only slightly higher than the 27.5 g associated with the deformable concrete overpack. The staff further notes the potential impact mitigating NAC-MPC design attributes, such as the relatively large VCC footprint and the small height-to-diameter aspect ratio. As a result, the staff has reasonable assurance that the calculated cask decelerations of 31.4 g and 27.5 g are representative and acceptable for subsequent application to the analysis of cask components.

SAR Section 11.2.12.3.1 reports a calculated pulse response duration of 30 milliseconds, which, together with the first-mode vibration frequency of 44 Hz, provides a basis for estimating a bounding dynamic load factor (DLF) of 1.61 for the support disk ligaments. The SAR applies this DLF to the LS-DYNA calculated cask deceleration of 27.5 g to arrive at a side-impact equivalent static load of 44.4 g for performing the quasi-static analysis of the support disks. Similarly, an equivalent static load of 34.2 g is determined for the canister lid region. Based on these static equivalent loads, the staff concurs with the SAR conclusion that it is acceptable to analyze the canister and basket assembly for the tipover accident represented by a side-impact quasi-static load of 45 g.

3.3.8.2 NAC-MPC System Components Structural Analysis

SAR Section 11.2.12.3 presents a symmetric one-half model of the upper portion of the canister and basket assembly for performing this structural evaluation. SAR Section 11.2.12.3.5 performs a sensitivity study which identifies the 45-degree orientation as bounding for the support disk impact evaluation, which is acceptable. On this basis, the staff concurs with the SAR conclusion that only the 45-degree orientation needs to be analyzed in detail.

SAR Tables 11.2.12.3-1 and -2 present, for 14 axial locations of the canister, the primary membrane and primary membrane-plus-bending stresses, respectively. The SAR performs stress evaluation per ASME Code, Section III, Subsection NB and Appendix F. In its February 24, 1999, submittal, NAC recalculated the stresses in the structural lid-to-canister shell weld by considering stress distribution over the area that the canister contacts in the VCC inner liner. In its February 27, 1999, submittal, NAC provided an additional evaluation of the weld by applying two separate design stress-reduction factors to the weld: (1) a 0.8 factor, per provision No. 5 of Interim Staff Guidance (ISG) No. 4, and (2) a 0.8 factor to reflect the fact that a full-penetration weld was substituted with a partial penetration weld. On the basis of the total design stress-reduction factor of 0.64 (0.8 x 0.8), NAC computed weld stress margins of 0.02 and 0.37 for the primary membrane and primary membrane-plus-bending stresses, respectively. For the TSC, these stress evaluations show a minimum stress margin of 0.02 for the primary membrane stress, which occurs in the structural lid-to-canister shell weld, and 0.1 for the primary membrane-plus-bending stress, which occurs in the TSC shell. These stress margins are acceptable.

SAR Tables 11.2.12.3-4 through -13 list the stress evaluation results for the five fuel basket support disks considered in the finite element model. The SAR performs stress evaluations per

the ASME Code, Section III, Subsection NG and Appendix F. The evaluation results show minimum margins of 1.51 and 0.21 for the primary membrane and primary membrane-plus-bending stresses, respectively. This is acceptable.

SAR Section 11.2.12.3.4 follows NUREG/CR-6322 to evaluate buckling of the support disk when subjected to in-plane compressive loads. The evaluation considers the maximum forces and moments in the ligaments. SAR Tables 11.2.12.3-14 through -18 list the buckling evaluation results on the basis of the interaction equations of the axial force and bending moment. The evaluation results show that the support disks meet the buckling criteria of NUREG/CR-6322.

3.3.9 Fuel Rod Rupture

The TSC is designed to remain leaktight to hold both intact and damaged fuel rods in a storage configuration. Because of this feature, the structural integrity of the fuel rod cladding is not considered in the design criteria for confinement of radioactive material under accident conditions. The spent fuel assemblies may undergo an axial impact of 19.95 g, in a 6-inch cask vertical drop accident, or a side impact of 44.4 g, in a cask tipover accident. Per NRC's ISG No. 3, the 10 CFR 72.122(l) regulation on fuel retrievability does not apply to post-accident recovery. Under normal and off-normal conditions, since the effects of pressure, thermal, and mechanical loadings on gross rupture of fuel rods are negligible, the staff has reasonable assurance that the spent fuel rods can readily be retrieved for further processing or disposal.

3.4 Natural Phenomenon Events

3.4.1 Flood

The design basis flood conditions of a 50-foot depth of water at a velocity of 15 feet/sec correspond to a hydrostatic pressure of 22 psig and a drag force of 21.7 kips on the VCC. The drag force is less than the minimum force of 113.4 kips required to cause the VCC to overturn and is not large enough to overcome the friction between the VCC and the concrete pad to cause the VCC to slide. At a hydrostatic pressure of 22 psig, the SAR reports a maximum primary membrane stress of 8.82 ksi and a maximum primary membrane-plus-bending stress of 19.18 ksi in the canister, which are below the stress intensity limits of 40.08 ksi and 60.12 ksi, respectively. On this basis, the staff agrees with the SAR Section 11.2.6 conclusion that the concrete cask will not overturn or slide and the NAC-MPC system will not suffer adverse structural consequences under the design basis flood conditions.

3.4.2 Tornado Wind and Tornado Driven Missiles

SAR Section 11.2.13 evaluates the structural performance of the NAC-MPC system under the design basis tornado wind and tornado driven missiles described in SAR Table 2.2-1. The VCC stability is evaluated in accordance with the ANSI/ASCE 7 wind pressure assumptions. Local damage to the VCC shell is assessed and the concrete shell capacity is evaluated with ACI 349 for a concrete compressive strength of 4,000 psi.

At a tornado wind velocity of 360 mph, the SAR calculates an effective pressure load of 23.6 kips applied on the VCC. This results in a safety factor of more than 7 against overturning and a friction coefficient of 0.12 required to prevent sliding. These results are acceptable.

The staff agrees with the SAR assessment that a detailed analysis of the TSC is not required for the impact of a 1-inch-diameter steel sphere because the missile cannot penetrate the VCC interior.

The SAR calculates a penetration depth of 5.68 inches due to a 275 lb, 8-inch-diameter armor piercing shell and determines that scabbing will not result in the 21-inch-thick concrete shell. For the same armor piercing shell impacting the VCC closure plate at 126 mph, the SAR estimates a perforation thickness of 0.645 inch. This is less than the plate thickness of 1.5 inches, and is acceptable.

Under the high energy deformable missile of 3960 lbs impacting the concrete cask at 126 mph, the SAR estimates a cask rotation of no more than 4.36 degrees. Considering an estimated impact force of 457.8 kips on the VCC top and the punching shear capacity, the staff concurs with the SAR conclusion that the concrete shell has sufficient capacity to withstand the high energy missile impact.

Based on the above evaluation, the staff concurs with the SAR conclusion that the design basis tornado wind pressure and tornado driven missiles are not capable of overturning the cask or penetrating the VCC. Therefore, the TSC confinement boundary remains intact.

3.4.3 Earthquake

SAR Section 11.2.2 evaluates the VCCs stability against sliding and tipover under the conditions of an earthquake. The evaluation assumes, at the top of a storage pad, a DBE motion of 0.25 g for the two horizontal components and 0.167 g for the vertical component. Considering the static equilibrium approach and the response combination criteria of ASCE 4, "Seismic Analysis of Safety-related Nuclear Structures," the SAR determines that a minimum horizontal component of 0.559 g is required to cause the VCC to tipover. This corresponds to a margin of 2.24 against tipover. The SAR also determines that a friction coefficient of 0.29 is adequate to prevent the VCC from sliding. On the basis of a commonly considered friction coefficient of 0.35 between the steel and concrete surfaces characteristic of the VCC base and the storage pad, the SAR calculates a margin of 1.21 against sliding. These margins are larger than the minimum of 1.1 per ANSI/ANS-57.9. As a result, the staff concurs with the SAR conclusion that the cask will not slide or tip over under the DBE condition.

3.4.4 Snow and Ice

The staff concurs with the SAR Section 3.4.4.2 conclusion that the maximum VCC snow load of 9,007 lbs is much smaller than the transfer cask live load of 135,000 lbs applicable to the TSC loading operation. Therefore, the snow and ice load has little structural consequence on the VCC.

3.5 Evaluation Findings

The NRC staff reviewed the SAR evaluation of the structural performance of the NAC-MPC for compliance with 10 CFR Part 72. The review considered the regulation, appropriate Regulatory Guides, applicable codes and standards, and accepted engineering practices. The NRC staff concludes that the NAC-MPC system will allow safe storage of spent fuel on the basis of the findings as follows.

F3.1 The SAR describes the SSCs important to safety in sufficient detail to enable an evaluation of the structural performance of the NAC-MPC designed to accommodate the combined loads of the normal, off-normal, and accident conditions and the natural phenomena events.

F3.2 The NAC-MPC is designed to allow ready retrieval of spent nuclear fuel for further processing or disposal. No normal or off-normal conditions analyzed will result in damage of the system to impair ready retrieval of the stored spent nuclear fuel.

F3.3 The NAC-MPC is designed and fabricated so that its structural performance is adequate for maintaining the spent nuclear fuel subcritical under the normal, off-normal, and accident conditions and the natural phenomenon events. Additional criticality evaluations are discussed in Section 6 of this SER.

F3.4 The SAR provides an acceptable evaluation of the structural integrity of the confinement boundaries of the NAC-MPC to demonstrate that the radioactive material will reasonably be confined under the normal, off-normal, and accident conditions and the natural phenomenon events.

F3.5 The SAR describes the materials that are used for SSCs important to safety and the suitability of those materials for their intended functions in sufficient detail to facilitate evaluation of their effectiveness.

F3.6 The design of the NAC-MPC and the selection of materials adequately protects the spent fuel cladding against degradation that might otherwise lead to gross rupture.

F3.7 The NAC-MPC employs noncombustible and heat-resistant materials which will help prevent the spread of radioactive material and maintain safety control functions.

F3.8 The materials that comprise the NAC-MPC will maintain their mechanical properties during all conditions of operation so the spent fuel can be readily retrieved without posing operational safety problems.

F3.9 The materials that comprise the NAC-MPC will maintain their mechanical properties during all conditions of operation so the spent fuel can be safely stored for a minimum of 20 years and maintenance can be conducted as required.

F3.10 The NAC-MPC employs materials that are compatible with wet spent fuel loading and unloading operations and facilities. These materials are not expected to degrade over time or react with one another during any conditions of storage.

4.0 Thermal Evaluation

The thermal review verifies that the cask and fuel material temperatures of the NAC-MPC system will remain within the allowable values or criteria for normal, off-normal, and accident conditions. This objective includes confirmation that the temperatures of the fuel cladding (fission product barrier) will be maintained throughout the storage period to protect the cladding against degradation which could lead to gross rupture. This portion of the review also confirms that the thermal design of the cask has been evaluated using acceptable analytical and/or testing methods.

4.1 Spent Fuel Cladding

The staff verified that the cladding temperatures for each fuel type proposed for storage are below the temperature limits which would preclude cladding damage that could lead to gross rupture.

The temperature limits for dry storage of stainless steel clad fuels are based on the technical justification provided in EPRI Report TR-106440. This report concluded that by placing a maximum temperature limit of 430°C (or 806°F) on the stainless steel cladding and by ensuring that cladding hoop stresses are less than the yield strength, the following types of degradation are not expected to occur: rapid stress rupture, creep failure, sensitization, and helium and strain rate embrittlement. Further, requiring an inert gas environment in the cask will preclude the occurrences of general and localized corrosion and stress corrosion cracking. As stated in the SRP for dry cask storage, NUREG-1536, Zircaloy clad fuel has a short-term temperature limit of 570°C (or 1058°F) per PNL-4835 research report and a long-term temperature limit of 340 °C per PNL-6364 research report. Thus, NAC established for both stainless steel and Zircaloy clad fuel a long-term temperature limit of 340°C (or 644°F) for normal conditions of storage and a short-term limit of 430°C for off-normal and hypothetical accident conditions. By establishing the short-term temperature limit of 430°C, NAC has effectively increased the margin of safety for Zircaloy clad fuel which could have used a short-term temperature limit of 570°C.

For cask unloading operations, the applicant considered the effect of cladding integrity by employing a two step cooling process of the hot spent fuel as described in Section 12 of the SAR, TS 3.1.7 and its associated Bases. The first step is to purge the cask of radioactive gases that may exist in the canister by purging with nitrogen for 10 minutes. After purging, an open water cooling system is connected to the drain line (inlet) and vent connection (outlet) with the discharge emptying into the spent fuel pool. During this cooling operation, inlet water temperature and flowrate are controlled to ensure that spent fuel cladding thermal stresses are maintained within acceptable limits and to ensure that the canister is not over-pressurized. The applicant further ensures over pressure protection of the canister during cooling by employing a check valve in the water inlet line that will restrict cooling water if the pressure in the canister gets too high and also uses a pressure relief valve. The staff reviewed the supporting reflooding calculation and finds that reasonable assurance has been provided to ensure that the spent fuel cladding will be adequately protected during unloading operations. In addition, the cladding temperature limit is further assured by placing a limit on the time that the canister is in the transfer cask (refer to Section 12 of the SAR, TS 3.1.10).

The staff reviewed the calculated values of temperature for all storage conditions to ensure that they met the 430°C temperature limit. The highest temperature that the cladding (Zircaloy or stainless steel) would experience is 638°F, as a result of a hypothetical fire accident condition. Since the calculational methods are conservative and there is a large margin between the calculated fire accident temperature and the temperature limit (i.e., 168°F), the staff concludes that the cladding would not be susceptible to the degradation mechanisms identified in the EPRI TR-106440 report and would remain intact for the license period.

4.2 Cask System Thermal Design

4.2.1 Design Criteria

The design criteria for the NAC-MPC storage cask have been formulated by the applicant to assure that public health and safety will be protected during the period that spent fuel is stored in the cask. These design criteria cover both the normal storage conditions for the 20-year approval period and postulated accidents that last a short time, such as a fire.

Section 4 of the SAR defines several primary design criteria for NAC-MPC components:

1) the spent fuel cladding temperature limit of 340°C (or 644°F) to prevent cladding degradation during long-term dry storage conditions for the NAC-MPC are based on research reports, PNL-6189 for Zircaloy clad spent fuel and EPRI TR-106440 for the stainless clad fuel;

2) the spent fuel cladding temperature limit of 430°C (or 806°F) to prevent cladding degradation during short-term storage conditions is based on EPRI TR-106440 for stainless steel clad spent fuel and is justified for Zircaloy cladding based on PNL-4835;

3) the design temperatures for the structural steel components of the NAC-MPC are based on the temperature limits provided in ASME Section II, Part D; other material design temperature limits are justified via consensus codes, military standards, or manufacturer's data; and

4) the thermal source term is based on fuel assembly types as noted in the TS which results in a maximum decay heat load of 12.5 kW per canister for intact spent fuel and 0.102 kW for an RFA (or 3.67 kW per canister).

4.2.2 Design Features

To provide adequate heat removal capability, the applicant designed the NAC-MPC system with the following features:

1) helium backfill gas for heat conduction which also provides an inert atmosphere for the fuel to prevent cladding oxidation and degradation;

2) relatively low decay heat load per assembly (i.e. < 0.35 kW per assembly); and

3) aluminum heat conduction elements for heat transfer from the fuel tubes to the canister shell.

The staff verified that all methods of heat transfer internal and external to the NAC-MPC are passive. The drawings in Section 1.5 of the SAR along with the material properties in Section 4.2 provide sufficient detail for the staff to perform an in-depth evaluation of the thermal performance of the entire package as required by 10 CFR 72.24(c)(3).

4.3 Thermal Load Specifications

The design basis fuel to be stored in the NAC-MPC is described in the TS for the Yankee Class intact spent fuel assemblies. The design basis spent fuel decay heat loading is 12.5 kW per canister for the intact fuel, which correlates to a heat loading of 0.347 kW per assembly for Zircaloy clad fuel. There may be up to 36 Zircaloy clad and/or stainless steel clad spent fuel assemblies per canister, as long as the canister weight limits are also met. The stainless steel clad spent fuel is limited to a decay heat load of 0.264 kW per assembly. Actual decay heat loads may be lower based on variances in burnup, decay time, enrichment, and amount of initial heavy metal in the fuel assembly. The Yankee Class fuels also vary in manufacturer (4), cladding (Zircaloy and stainless steel), configuration (2), initial fill gas pressure (0 to 315 psi), and other minor differences that are identified in the TS. The limits for decay heat, burnup, cooling time, and enrichment are also specified in the TS and Table 2.1-1 of the SAR. The axial power distribution for Yankee Class fuel is shown in Figure 4.4.1.1-3 of the SAR. The decay heat load will decrease with time over the storage period.

The NAC-MPC contents may include RFAs containing intact or damaged spent fuel rods or fuel debris. The RFA contents are contained in half-inch diameter stainless steel tubing which is arranged in an 8 x 8 array which is the same nominal size as the standard Yankee Class spent fuel assembly. The fuel assembly limits are identified in the TS for both the intact and reconfigured fuel assemblies.

The staff reviewed and confirmed via analysis the decay heat loads for each fuel design identified in the TS and in Table 2.1-1 of the SAR. The staff also verified that the design basis decay heat load is bounding through independent analysis providing reasonable assurance that the design basis decay heat load was determined properly. For the RFA, the damaged fuel shall meet the same specifications as the intact fuel except for a limitation on decay heat of 0.102 kW and other size and weight restrictions identified in the TS.

The thermal loads are different for the normal storage conditions than for the accident conditions, such as fire. The difference with the thermal loads occurs at the surface of the canister or cask. The application of the surface thermal loads will be for a short time during an accident, while the thermal loads from the fuel assembly decay heat are applied continuously during normal storage conditions. The decay heat load during an accident will be the same as for the normal storage condition at the time of the accident.

4.3.1 Normal Storage Conditions

The external environment for normal storage conditions are described in the SAR. The applicant evaluated the cask for conditions with an ambient temperature of 75°F and solar

insolance in a 12-hour period of 2950 BTU/ft^2 and 1475 BTU/ft^2 for horizontal flat and curved surfaces, respectively.

4.3.2 Off-Normal Conditions

The external environment for off-normal storage conditions associated with the thermal evaluation are described in Table 4.1-1 and Sections 11.1.1, and 11.1.4 of the SAR. The off-normal events analyzed include half of cask inlets blocked, severe environmental heat of 100° F, and severe environmental cold of -40° F.

The applicant evaluated the NAC-MPC for conditions associated with half of the cask inlets blocked, including an environmental temperature of 75°I and a 12-hour insolation period of 2950 BTU/ft^2 and 1475 BTU/ft^2 for horizontal flat and curved surfaces, respectively.

The applicant evaluated the NAC-MPC for a severe environmental heat of 100°F and a 12-hour insolation period of 2950 BTU/ft^2 and 1475 BTU/ft^2 for horizontal flat and curved surfaces, respectively. Also, the maximum decay heat of 12.5 kW was modeled as identified in Section 11.1.4 of the SAR.

The applicant also evaluated the NAC-MPC for conditions with ambient temperatures of -40°F, no solar insolation, and applied the maximum decay heat of 12.5 kW, as described in Section 11.1.4 of the SAR . The staff concurs with this approach since the largest radial thermal gradient would exist with the maximum decay heat load and thus produce the largest thermal stresses. Also, since the material of the canister is a ductile stainless steel, it would not be susceptible to brittle fracture associated with the colder temperatures.

4.3.3 Accident Conditions

Three thermal accidents are postulated and individually evaluated for the NAC-MPC. They include a fire, an all air inlets and outlets blocked event, and an extreme environmental temperature of 125°F event.

The thermal accident postulated for the NAC-MPC is described in Section 11.2.5 of the SAR. A fire with an average flame temperature of 1475°F and duration of 8 minutes is postulated from the spillage and ignition of 50 gallons of combustible transporter fuel. The fire is assumed to spread along the ground and heat the air as it enters the cask. Solar insolation is applied during the fire since the fire is only assumed to occur at the base of the cask and a heat load of 12.5 kW is considered. The initial temperature distribution of the transient is based on the normal storage conditions. Following the fire, the cask is allowed to cool for 60 minutes using the boundary conditions corresponding to the normal conditions of storage.

A full blockage of all air inlets and outlets on the cask is described in Section 11.2.8 of the SAR. The event initiates at normal conditions with an ambient air temperature of 75°F and is postulated to result from a greater than DBE or landside.

An extreme environmental heat of 125°F is analyzed for the NAC-MPC with a maximum decay heat of 12.5 kW and a 12-hour insolation period of 2950 BTU/ft^2 and 1475 BTU/ft^2 for horizontal flat and curved surfaces, respectively.

4.3.4 Transfer Conditions

The applicant analyzed the temperature rise of the transfer cask components and canister contents beginning from the placement of spent fuel in the transfer cask and concluding with the placement of the loaded canister in the VCC. This analysis is composed of three steps: (1) wet loading and draining of the canister, (2) vacuum drying, and (3) helium filling, canister final sealing, and placement in the VCC. During the initial wet loading operation, the applicants consideration of the minimum time to boil avoids the potential for uncontrolled pressure increases due to the water boiling in the cask during transfer. By determining a bounding heat-up rate based on maximum heat load and minimum volume, the applicant established the time before water in the canister would boil as a function of the initial temperature of the NAC-MPC contents when removed from the pool. In addition, the applicant analyzed the temperature rise of the canister contents and transfer cask components during the vacuum drying operations and subsequent helium filling, sealing, and transfer to the VCC. The results of the analysis justify a time period for completion of the transfer operations with all components being below their allowable temperature limits and is shown in Figure 4.4.3-5 of the SAR. Operational implementation of the transfer temperature limits is accomplished by establishing heat-up rates for each of the transfer conditions of water heat up, vacuum drying, and helium filling with no external cooling. This permits flexibility between the three transfer conditions such that the time periods for each condition can be adjusted from the analyzed time period while providing assurance that the temperature limits of the components are not exceeded during transfer operations. The staff reviewed the associated calculation, procedures, and TS, including the required actions, and verified that the application of these heat-up rates is bounded by the component temperature limits.

4.4 Model Specification

4.4.1 Configuration

The analytical model for the thermal design of the NAC-MPC system was developed using the industry standard ANSYS computer code. The computational fluid dynamics modeling that was performed utilized ANSYS FLOTRAN. Transport of heat from the fuel assemblies to the outside environment is analyzed in terms of four interdependent thermal models. The first model considers air flow within the VCC, as well as, temperature distribution in the canister shell and VCC. The second model considers heat transport within the canister. The third model is used to determine the effective conductivity of the fuel. The fourth model is used to determine the effective conductivities of the tube wall and Boral plate. Two other models are used to evaluate the thermal design of the cask with canister and to evaluate the RFA. Natural convection of air in the VCC, in addition to radiative heat transfer from the canister surface, cools the spent fuel cladding and storage cask components below their temperature limits.

4.4.1.1 Air Flow and Concrete Cask Model

The air flow and concrete cask model consists of the canister shell, steel inner liner of the concrete cask, concrete, air inlet and outlet, and annulus region. The canister shell is included in the model to apply the design decay heat load of 12.5 kW which is applied in the form heat flux based on the axial power distribution of the fuel. Cooling of the canister shell is by natural convection in the annulus region between the canister and concrete cask and by radiation heat transfer between the canister shell and the VCC inner liner. Heat is also dissipated through the concrete and dissipated to the surroundings by natural convection and by radiation heat transfer. Since the four air inlets and four air outlets are symmetrical about the axis of the cask, a two-dimensional axisymmetrical air flow and concrete cask finite element model was used, as shown on Figure 4.4.1.1-2 of the SAR. Insolation is also ac ded to the outer cask surface and is averaged over a 12-hour period. This thermal system, including mass, momentum, and energy was analyzed using ANSYS FLOTRAN. Also, the applicant performed an accuracy check of the numerical solution to this model in three areas. First, the global convergence of the iteration process for the nonlinear system was shown to converge after 10,000 iterations (refer to SAR Figure 4.4.1.1-4) and all results were presented in the converged state. Second, the number and size of the elements were demonstrated to produce only a 3% improvement for heat load for a 28% increase in the number of elements. Third, the energy balance for the ratio of total heat output to total heat input was demonstrated to be within 1.5%.

The applicant considered the thermal interaction among casks in an array. The view factor between the most adversely located system in the middle of the array with all of the other neighboring casks is determined using a classical formula for radiation exchange between surfaces. This view factor for the cask surface is used in the air flow and concrete cask model.

4.4.1.2 Canister Model

The canister model as shown in Figure 4.4.1.2-1 includes the canister shell, including the lids and bottom plate; and the internals including the fuel assemblies, fuel tubes, support disks, heat transfer disks, and helium. A three-dimensional model was used, and since the canister is symmetric, only half of it was modeled. Conduction heat transfer is modeled in the axial direction of the active fuel region. Conduction was not modeled in the axial direction at the ends of the fuel bundle, which adds to the conservatism of the model. The outside surfaces of the canister structural lid and bottom plate are modeled as adiabatic, again adding conservatism to the model. The fuel assemblies and fuel tubes are assumed in the model to be concentrically centered in the support/heat transfer disk slots with gaps existing all around. Likewise, the fuel bundle is assumed to be centered within the canister shell with a gap existing between the support/heat transfer disks and the canister shell. The size of the gaps used in the model were demonstrated to be bounding of the nominal gaps adjusted for thermal expansion and fabrication tolerances. Radiation heat transfer was modeled across all of the aforementioned gaps, as well as between the exterior surface of the fuel tubes and the canister shell and between the fuel region and the adjacent surfaces of the shield lid and canister bottom plate. Radiation heat transfer was not considered between the exterior surfaces of the fuel tubes, which is conservative. An effective conductivity was determined via a separate ANSYS model for the region inside of the fuel tubes, including the helium gas. Additionally, an effective conductivity was determined via another separate ANSYS model for

the fuel tube (including Boral plate), including helium gaps on both sides of the Boral plate and the helium gap between the fuel tube exterior surface and the heat transfer/structural disks. The decay heat load of 12.5 kW was applied in the active fuel region based on the axial power distribution shown in Figure 4.4.1.1-3 of the SAR.

4.4.1.3 Fuel Model

A full cross-section of the fuel assembly is modeled to determine the effective conductivity of the fuel. The model includes the fuel pellets, helium gas in the fuel rod, cladding, and helium between the fuel rods. Zircaloy is used for the cladding material since it is assumed to produce higher fuel cladding temperatures than the stainless steel, based primarily on Zircaloy having a lower thermal conductivity. Radiation heat transfer is modeled between the fuel rods and between the fuel rods and the inside surface of the fuel tube. The design heat load of 12.5 kW is applied to the fuel pellets in the form of a volumetric heat generation.
Figure 4.4.1.3-1 shows the two-dimensional fuel model.

4.4.1.4 Fuel Tube Model

A two-dimensional fuel tube model is used to determine the effective conductivity of the fuel tube and Boral plate as it varies with temperature. The model includes the fuel tube, the Boral plate, helium gaps on both sides of the Boral plate, and the helium gap between the fuel tube exterior surface and the heat transfer/structural disks. The model consists of conduction through all layers with radiation only at the helium gaps. The design basis heat load of 12.5 kW is applied as a heat flux to the inside surface of the fuel tube. ANSYS results are used to calculate the maximum temperature at the inside surface of the fuel tube. Figure 4.4.1.4-1 in the SAR shows the two-dimensional fuel tube model.

4.4.1.5 Transfer Cask and Canister Model

A three-dimensional model is used to determine the transient temperature distribution for the fuel cladding, transfer cask, and canister, including internal components, for the transfer condition. The model of the canister, discussed previously, is used with the transfer cask added. The model is a one quarter segment of the canister and transfer cask assembly. Convection and radiation heat transfer are considered at the surfaces of the transfer cask and on top of the canister lid. An adiabatic boundary is assumed for the bottom of the transfer cask. The canister is assumed to be concentric with the transfer cask. Air gaps are modeled in two places: for the space between canister shell and the inner shell of the transfer cask and the space between transfer cask inner shell and lead. Radiation heat transfer is modeled across these gaps. A volumetric heat generation, determined from the maximum heat load of 12.5 kW, is applied to the active fuel region utilizing the axial power distribution from Figure 4.4.1.1-3 of the SAR. The transfer cask and canister model is shown on Figure 4.4.1.5-1 of the SAR.

4.4.1.6 Reconfigured Fuel Assembly Model

A two-dimensional, one quarter, cross-sectional model of the RFA is used to determine the maximum temperatures of the RFA components, including the fuel cladding. Each RFA stainless steel rod is modeled to contain a helium gas gap and a solid fuel rod with the thermal

conductivity of UO_2. The use of a solid fuel rod with the thermal conductivity of UO_2 is used since its conductivity is less than Zircaloy and would produce a conservatively higher cladding temperature. All gases are considered to be helium. Radiation heat transfer is modeled between the RFA tubes and from the RFA tubes to the RFA shell casing. Based on a design heat load of 0.0016 kW per RFA pin, a volumetric heat generation is applied to the fuel rod. Also, a peaking factor of 1.15 is applied. The RFA model is shown on Figure 4.4.1.6-1 of the SAR.

4.4.2 Material Properties

The material properties used in the thermal analysis of the storage cask system are listed in Section 4.2 of the SAR. The applicant provided material compositions and therma properties for all components used in the calculational model. The material properties given reflect the accepted values of the thermal properties of the materials specified for the construction of the cask. For homogenized materials, such as described in the fuel tube model, the applicant adequately described the manner in which the effective thermal properties were calculated.

4.4.3 Boundary Conditions

The boundary conditions for the NAC-MPC include the design basis decay heat of 12.5 kW and the external conditions on the cask surface. The distribution of the decay heat load is based on the axial power distribution curve shown on Figure 4.4.1.1-3 of the SAR. From the curve, the peak power factor for the fuel is 1.15, which is also discussed in SAR Section 5.2.3.

The boundary conditions depend on the environment surrounding the cask. Three conditions are considered for the TSC. The first includes the conditions set forth for normal storage. The second case considers off-normal operation, like severe environmental temperatures. The third case considers the effect of accidents, such as a fire, on the thermal performance of the cask. A summary of the thermal design conditions for storage, including environmental temperature, insolation, and vent operation are provided in Table 4.1-1, except for the fire accident which is described in Section 11.2.5 of the SAR.

4.4.3.1 Normal Storage Conditions

The applicant evaluated the cask for conditions with an ambient temperature of 75°F and a 12-hour insolation period of 2950 BTU/ft^2 and 1475 BTU/ft^2 for horizontal flat and curved surfaces, respectively. VCC inlets and outlets were assumed to be free from any blockage and the design basis heat load of 12.5 kW was applied.

4.4.3.2 Off-Normal Storage Conditions

The applicant evaluated the cask for the severe heat condition of an ambient temperature of 100°F and a 12-hour insolation period of 2950 BTU/ft^2 and 1475 BTU/ft^2 for horizontal flat and curved surfaces, respectively. The design basis heat load of 12.5 kW was also applied. This off-normal condition is described in Section 11.1.4 of the SAR.

The second off-normal condition evaluated a severe cold ambient temperature of -40°F, no solar insolation, but the design basis heat load of 12.5 kW was applied. The decay heat load

was applied to this condition since brittle fracture of the canister is not a concern due to the ductility of the stainless steel and the maximum thermal stress would be associated with the largest decay heat. This off-normal condition is described in Section 11.1.4 of the SAR.

The third off-normal condition evaluated the cask for half of the air inlets blocked and is described in Section 11.1.1 of the SAR. The ambient temperature used for this condition was 75°F, solar insolation was applied as for the severe heat condition, and the design basis heat load of 12.5 kW was applied.

4.4.3.3 Accident Conditions

Three accident conditions were analyzed by the applicant: an extreme heat condition, a blockage of all cask air inlets and outlets, and a fire.

The accident extreme heat condition of 125°F ambient was analyzed at steady-state conditions as described in Section 11.2.10 of the SAR. Full solar insolation was applied in addition to the design basis heat load of 12.5 kW.

The second accident condition analyzed blockage of all cask inlets and outlets as described in Section 11.2.8 of the SAR. Full solar insolation was applied in addition to the design basis heat load of 12.5 kW. As an added conservatism, no heat loss from the external surface of the cask was considered which was done to also bound the buried cask scenario. However, a buried cask would not consider solar insolation as was modeled here.

The third accident condition postulated a fire as described in Section 11.2.5 of the SAR. A fire resulting from a spillage and ignition of 50 gallons of transporter diesel fuel with an average flame temperature of 1475°F is hypothesized to last 8 minutes. Since the cask air inlets are at grade elevation, the air entering the cask is heated by the postulated fire. At the start of the transient, the ambient air temperature is changed instantaneously to 1475°F, and the air inlet surfaces, up to an elevation 10 inches above the base, are changed to 1475°F. The initial temperature distribution of the transient is based on the normal storage conditions, including a decay heat load of 12.5 kW and solar insolation applied to the top and sides of the cask. Following the fire, the external conditions for normal operating steady-state conditions are applied and this cooldown phase is then analyzed for 60 more minutes.

4.5 Thermal Analysis

4.5.1 Computer Programs

The complete thermal analysis was performed by the applicant using the industry standard ANSYS finite element modeling package and its associated computational fluid dynamics code, FLOTRAN. ANSYS is capable of general three-dimensional steady-state and transient calculations. Based on the SAR drawings and the thermal property information contained in the SAR, the staff determined that sufficient information was available to perform confirmatory analysis. The staff reviewed selected applicant calculations to confirm that the modeling was performed in accordance with the drawings and the boundary conditions contained in the SAR and that the calculations agreed with the conclusions presented in the SAR. The staff performed confirmatory calculations to verify the heat loads associated with the various

proposed contents were bounded by the design basis heat load. Thus, the staff has reasonable assurance that the NAC-MPC spent fuel storage system provides adequate heat removal capacity without an active cooling system, as required by 10 CFR 72.236(f).

4.5.2 Temperature Calculations

4.5.2.1 Normal Storage Conditions

The NAC-MPC system has been analyzed to determine the temperature distribution under long-term normal storage conditions. The canister has been considered to be loaded at design basis maximum heat loads with Yankee Class spent fuel assemblies, as appropriate. The systems are considered to be arranged in an ISFSI array and subjected to design basis normal ambient conditions with insolation. The maximum allowable temperatures of the components important to safety are listed in Table 4.1-3 of the SAR. Low temperature conditions were also considered.

The table below summarizes the applicant's calculated temperatures of key components associated with the storage of intact spent fuel for various environmental conditions.

NAC-MPC Cask Component	Normal Conditions [°F]		Off-Normal & Accident Conditions [°F]				
	Normal Conditions	Allowable (Long-term)	Blockage of Half of Air Inlets (Off-Normal)	Extreme Environmental Conditions (Accident)	Fire (Accident)	Helium Transfer (at 66 hours)	Allowable (Short-term)
Fuel Cladding	563	644	565	607	638	597	806
Aluminum Heat Transfer Disk	527	650	529	574	602	569	700
Stainless Steel Support Disk	529	650	531	575	604	570	800
Canister Shell	319	800	318	372	394	430	800
Concrete Liner	165	700	169	229	NA	NA	700
Concrete	165 (local) 133 (bulk)	200 (local) 150 (bulk)	168	228	144	NA	350
Lead	NA	NA	NA	NA	NA	191	600
Neutron Shield	NA	NA	NA	NA	NA	188	300
Transfer Cask Shell	NA	NA	NA	NA	NA	237	700

The applicant's calculated maximum temperatures for the dry storage of reconfigured fuel assemblies is presented in Table 4.4.3-4 of the SAR and are shown below for completeness.

NAC-MPC RFA Component	RFA Normal Conditions [°F]		RFA Off-Normal & Accident Conditions [°F]		
	Normal Conditions	Allowable (Long-term)	Off-Normal & Accident	Helium Transfer (at 66 hours)	Allowable (Short-term)
PWR Fuel Tube	540	800	580	715	800
RFA Shell Casing	543	800	583	718	800
RFA Tube	563	800	602	734	800
Fuel Rod Cladding	563	644	602	734	806

As can be seen from the above tables for the intact and RFA spent fuel, all of the calculated component temperatures are below their allowable temperature.

4.5.2.2 Off-Normal Conditions

The off-normal event considering an environmental temperature of 100°F for a duration sufficient to reach thermal equilibrium was evaluated by the applicant. The evaluation was performed with design basis fuel with maximum decay heat. The 100°F environmental temperature was applied with full solar insolation. All of the off-normal temperatures were below the short-term design basis temperatures.

The off-normal event considering an environmental temperature of -40°F, design basis decay heat, and no solar insolation for a duration sufficient to reach thermal equilibrium was evaluated by the applicant. The use of the maximum decay heat load produces the maximum thermal gradient with respect to the calculation of thermal stresses for this condition. Also, the structural evaluation in Section 3.4.5 of the SAR demonstrated that there was no reduction in the performance of the NAC-MPC system after 20 years of storage with respect to brittle fracture.

Analysis of the off-normal event of blockage of half of the air inlets demonstrates that the cask has an adequate air supply since the results indicate only a few degree temperature rise above normal conditions.

Based on these analyses and review, the staff has reasonable assurance that the off-normal temperatures do not affect the safe operation of the NAC-MPC.

4.5.2.3 Accident Conditions

The extreme environmental conditions considering an environmental temperature of 125°F for a duration sufficient to reach thermal equilibrium were evaluated by the applicant. The evaluation was performed with design basis fuel and with maximum decay heat. The 125°F environmental temperature was applied with full solar insolation. All of the extreme environmental temperatures were below the short-term design basis.

Based on these analyses and review, the staff has reasonable assurance that the extreme environmental conditions do not affect the safe operation of the NAC-MPC.

The applicant analyzed a fire accident on the NAC-MPC system using the conditions previously specified in SER Section 4.4.3.3. The peak temperatures of the key cask components due to an 8-minute fire with a maximum decay heat are shown in the table in SER Section 4.5.2.1. The staff verified, via independent calculation, using the method documented in NUREG-0360 that the duration of the fire is conservative. The initial temperatures are based on the normal storage conditions and an incident solar heat flux based on the specified insolation averaged over 24 hours. The components inside the canister were not modeled directly. Instead, the temperature of the canister shell was determined from the air flow and concrete cask model. The temperature of the canister's internal components was then determined by adding the difference in canister shell temperature (between the normal and fire accident) to the maximum normal temperature of the canister internal components. This method results in a bounding cladding temperature that is independent of post fire cooling time scenarios. All of the fire accident temperatures were below the short-term design basis temperatures. Based on these analyses and review, the staff has reasonable assurance that the cladding integrity will not be compromised during the postulated fire.

The applicant analyzed the effect of blocking all the air inlets and outlets. This event was analyzed to determine the minimum time for reaching material temperature limits. The applicant was conservative in the assumptions for setting up the problem. Not only were adiabatic conditions established at the cask surface, simulating burial in a landside, but full solar insolation was also applied. The design basis heat load was also used. The results from this unlikely scenario indicate that the cask should not be deprived of air flow for more than 45.7 hours, otherwise the concrete or spent fuel cladding may exceed their temperature limit. The staff reviewed the associated operating procedures and TS to ensure that this design basis time limit would not be exceeded when air flow to the cask is lost.

4.5.3 Pressure Analysis

4.5.3.1 Normal Conditions of Storage

The applicant determined the pressure in the TSC based on the average cavity gas temperature of 450°F and the normal storage conditions. The canister is sealed and backfilled with 0.0 psig of helium at 70°F. During normal operating conditions, a conservative estimation of the design pressure was based on the calculation of the total moles of gas available for pressurizing the canister. The moles of gas available for pressurization of the canister due to decay heating include moles: from backfilling the canister with helium, from 3% of the spent fuel leaking its initial helium fill gas at a pressure of 315 psig, and from 30% of the fission gas produced from the assumed 3% leaking spent fuel rods. Under these conditions, and with the average cavity gas temperature of 450°F , the applicant determined a design pressure of 7.9 psig. Therefore, this determination of design pressure justifies the test pressure of 50 psig (determined for future transport considerations) mentioned in Section 9.1.2 of the SAR, since the ASME Code for Class 1 components only requires a pneumatic test at a factor of 1.2 of the design pressure.

4.5.3.2 Off-Normal Conditions

The applicant did not provide an explicit derivation of the off-normal pressure in the SAR. However, the staff considers this calculation straight forward in that it is derived from the methods employed in determining the design pressure, except that the fraction of damaged fuel rods is changed from 3% to 10% and a bulk average gas temperature coinciding with off-normal operation is used. This pressure would be used in determining stresses for off-normal loading combinations.

4.5.3.3 Accident Conditions

The applicant determined the pressure in the TSC based on the average cavity gas temperature of 650°F, which is conservative since the maximum fuel cladding temperature during the fire is only 638°F. Using the gas temperature of 650°F, an assumed spent fuel rod cladding failure of 100%, and the methodology from the design pressure calculation, the applicant determined an accident maximum canister pressure of 43.3 psig. The applicant increased this pressure to 55 psig to bound all pressurization scenarios, including the canister reflooding operation, and used this value in the calculation of stresses due to internal canister pressurization as shown in Tables 11.2.1-1 and 11.2.1-2 of the SAR.

4.5.4 Confirmatory Analysis

The confirmatory analysis of the NAC-MPC storage cask SAR can be divided into seven categories: (1) review the models used in the analyses, (2) review the material properties used in the analyses, (3) review the boundary conditions and assumptions, (4) perform independent analyses, (5) review selected applicant calculation packages, (6) compare the results of the analyses with the applicant's design criteria, and (7) assure that the applicant's design criteria will satisfy the regulatory acceptance criteria and regulatory requirements.

The staff reviewed the models used by the applicant in the thermal analyses. The code inputs in the calculation packages were checked for consistency to confirm that the applicant used the appropriate material properties and boundary conditions where required. The engineering drawings were also consulted to verify that proper geometry dimensions were translated to the code model. The applicant justified their modeled gap values for consideration of fabrication tolerances, in addition to thermal growths. The material properties presented in the SAR were reviewed to verify that they were appropriately referenced and used conservatively. In addition, the staff performed a confirmatory analysis of the thermal heat loads to ensure that they were bounded by the design basis heat load. Also, consistent with the transportation analyses under separate staff review, the accident pressure used in the stress calculations was changed from 35 psig to 55 psig.

The staff has determined that the thermal SSCs important to safety are described in sufficient detail in Sections 1 and 4 of the SAR to enable an evaluation of their effectiveness. Based on the applicant's analyses, there is reasonable assurance that the NAC-MPC is designed with a heat-removal capability having testability and reliability consistent with its importance to safety.

Based on the applicant's analyses, there is reasonable assurance that the NAC-MPC system provides adequate heat removal capacity without active cooling systems. The staff also has

reasonable assurance that the spent fuel cladding will be protected against degradation that leads to gross ruptures by maintaining the clad temperature below the allowable criteria and by providing an inert environment in the cask cavity, thus assuring that the fuel can be readily retrieved for future processing or disposal without significant safety problems.

The staff has further concluded that the design of the heat removal system of the NAC-MPC cask is in compliance with 10 CFR Part 72 and that the applicable design and acceptance criteria have been satisfied. The evaluation of the thermal system design provides reasonable assurance that the NAC-MPC will enable safe storage of spent fuel. This finding is based on a review which considered the requirements of 10 CFR Part 72, appropriate regulatory guides, applicable codes and standards, and accepted practices.

4.6 Evaluation Findings

F4.1 Thermal SSCs important to safety are described in sufficient detail in Sections 1 and 4 of the SAR to enable an evaluation of their effectiveness [10 CFR 72.24(c)(3)].

F4.2 The staff has reasonable assurance that the spent fuel cladding will be protected against degradation that leads to gross ruptures by maintaining the clad temperature below maximum allowable limits and by providing an inert environment in the cask cavity [10 CFR 72.122(h)(1)].

F4.3 Through the analysis, staff developed reasonable assurance that the NAC-MPC system is designed with a heat-removal capability having testability and reliability consistent with its importance to safety [10 CFR 72.128(a)(4)].

F4.4 By analysis, the staff has reasonable assurance that the decay heat loads were determined appropriately and accurately reflect the burnup, cooling times, and initial enrichments specified [10 CFR 72.122].

F4.5 By analysis, the staff has reasonable assurance that the NAC-MPC system provides adequate heat removal capacity without active cooling systems [10 CFR 72.236(f)].

F4.6 By analysis, the staff has reasonable assurance that the temperatures of the cask components and the cask pressures under normal and accident conditions were determined correctly [10 CFR 72.122].

5.0 SHIELDING EVALUATION

The purpose of the shielding evaluation is to determine whether the NAC-MPC shielding features provide adequate protection against direct radiation from cask contents. The regulatory requirements for providing adequate radiation protection to licensee personnel and members of the public include 10 CFR Part 20, 10 CFR 72.104(a), 72.106(b), 72.212(b), and 72.236(d). Because 10 CFR Part 72 dose requirements for members of the public include direct radiation, effluent releases, and radiation from other uranium fuel-cycle operations, an overall assessment of compliance with these regulatory limits is evaluated in Section 10 (Radiation Protection) of the SER.

5.1 Shielding Design Description

5.1.1 Shielding Design Criteria

Section 2 of the SAR specifies the principal design criteria of the NAC-MPC. Section 5.1 of the SAR provides the design criteria for the surface dose rates of the cask. The design basis fuel for the NAC-MPC has been determined to be Combustion Engineering (CE) Type A fuel assemblies, with a maximum burnup of 36,000 MWD/MTU with a minimum cool time of 8 years.

The average surface dose rate criteria for the NAC-MPC at the concrete cask side wall shall be less than 50 mrem/hr , at the concrete cask top lid less than 35 mrem/hr, and at the concrete cask air inlet/outlet less than 100 mrem/hr. The overall design criteria for the NAC-MPC are the regulatory dose limits in 10 CFR Part 20, 10 CFR 72.104(a), and 10 CFR 72.106(b).

5.1.2 Shielding Design Features

The NAC-MPC is designed to provide both gamma and neutron shielding. The principal components of the NAC-MPC system are the concrete storage cask, TSC, and transfer cask. The TSC is designed to be transported in the NAC-STC transport cask. The TSC, fabricated from Type 304L stainless steel, contains a basket that will hold up to 36 Yankee Class spent fuel assemblies.

The transfer cask, used to hold the canister during fuel loading activities and to transfer the canister to the storage cask, has a multi-wall radial shield comprised of 1.91 cm (0.75 in) of carbon steel, 8.89 cm (3.5 in) of lead, 5.08 cm (2 in) of solid borated polymer (NS-4-FR), and 3.18 cm (1.25 in) of carbon steel. An additional 1.6 cm (0.625 in) of stainless steel shielding is provided, radially, by the canister shell. Gamma shielding is provided primarily by the steel and lead layers. The NS-4-FR provides the neutron shielding. The transfer cask bottom is a solid section of 24.13 cm (9.50 in) of carbon steel. The top of the transfer cask is open but shielding is provided by the stainless steel canister shield and structural lids which are 12.70 cm (5 in) and 7.62 cm (3 in) thick, respectively. Additionally, there is a 12.70 cm (5 in) carbon steel temporary shield which may be used during welding, draining, drying, and helium backfill operations.

The VCC is the storage overpack for the TSC. The storage cask is 53.34 cm (21 in) of reinforced concrete (Type II Portland cement) structure with a 8.89 cm (3.5 in) thick carbon steel inner liner. The concrete wall and steel liner provide the neutron and gamma radiation shielding to reduce the average contact dose rate to less than 50 millirem per hour for the design basis fuel. An additional 1.6 cm (0.625 in) of stainless steel shielding is provided, radially, by the canister shell. The storage cask top is comprised of the stainless steel canister lids 20.32 cm (8 in) thick, a shield plug containing 2.54 cm (1 in) NS-4-FR and 10.48 cm (4.125 in) of carbon steel, and a carbon steel lid 3.81 cm (1.5 in) thick. The storage cask bottom, which is composed of 2.54 cm (1 in) of stainless steel from the canister bottom plate, 5.08 cm (2 in) of carbon steel from the pedestal plate, and 2.54 cm (1 in) of carbon steel cask base plate, will rest on a concrete pad.

The staff evaluated the NAC-MPC shielding design features and criteria and found them to be acceptable. The SAR analysis provides reasonable assurance that the shielding design features and criteria can meet the regulatory requirements in 10 CFR Part 20, 10 CFR 72.104(a), and 10 CFR 72.106(b). Cask surface dose limits are included in TS 3.2.1.

5.2 Radiation Source Definition

Section 2.1 of the SAR describes the spent fuel to be stored in the NAC-MPC and identifies the type of fuel which has been determined to be bounding for the shielding evaluation. Table 2.1-1 contains the parameters for the Yankee Class fuel assemblies. Section 5.2 of the SAR presents the source specification for the Yankee Class fuel assemblies.

The SAS2H module of the SCALE 4.3 code package for the PC (ORNL) was used to generate gamma and neutron source terms. The 27 energy group ENDF/B-IV neutron cross section library was used in the source term calculations. The fuel assembly hardware source term is calculated by light element transmutation using the incore neutron flux produced by the SAS2H model. The fuel hardware is assumed to be Type 304 stainless steel that has a ^{59}Co impurity level of 1.2 g/kg and some minor contaminants from ^{59}Ni and ^{58}Fe.

The NAC-MPC is designed to store up to 36 Yankee Class fuel manufactured by Combustion Engineering (CE), Exxon, United Nuclear (UN), and Westinghouse (WE) in two fuel rod configurations. CE, Exxon, and UN fuel assemblies are 16 x 16 Type A and Type B with Zircaloy cladding. WE fuel assemblies are 18 x 18 stainless steel clad Type A and Type B.

The CE 16 x 16 Type A fuel assembly with a minimum initial enrichment of 3.7 w/o U-235, burnup of 36,000 MWD/MTU and 8 years cool time was established as the Yankee Class design basis fuel assembly for the shielding evaluations. CE, UN, and WE Yankee Class fuel assemblies with a maximum burnup of 32,000 MWD/MTU with minimum cool times of 7.0, 7.1, and 21.0 years, respectively, may also be loaded in the NAC-MPC. Exxon fuel at 36,000 MWD/MTU requires a minimum cool time of 9 years for assemblies containing Zircaloy fuel region hardware and 16 years for assemblies containing stainless steel fuel region hardware.

The staff performed a confirmatory analysis of the design basis gamma and neutron source terms for the Zircaloy clad and stainless steel clad fuels. Staff used SAS2H and ORIGEN-S of the SCALE-4.4 computer code. The staff reviewed the fuel parameters listed in SAR tables 2.1-1 and 2.1-2 and has reasonable assurance that the design basis gamma and neutron source terms are adequate for the shielding analysis.

5.3 Model Specification

Section 5.3 of the SAR provides the model specifications for the shielding evaluation. NAC used the one-dimensional, SAS1, and three-dimensional, SAS4, models in the shielding evaluation for the NAC-MPC. The SAS1 radial and axial models were used to estimate the peak and average dose rates on the sides, top and bottom of the storage and transfer casks. SAS1 was also used to determine the minimum cool time necessary for fuel with a burnup of 36,000 MWD/MTU.

The SAS4 three-dimensional model was used to estimate the dose profiles at the surfaces of the cask and at streaming paths such as the storage cask inlets and outlets, and the canister vent and drain ports. SAS4 uses adjoint discrete ordinates and Monte Carlo methods in solving the shielding problem. Since SAS4 requires model symmetry at the fuel midpoint, two models are created for each cask, a top and bottom model. Radial biasing is performed to estimate the dose rates of the sides of the cask. Dose rates on the top and bottom surfaces of the cask are estimated by axial biasing.

SKYSHINE-III Version 5.0.0 was used to calculate the controlled area boundary dose for a proposed ISFSI of 16 NAC-MPC casks (SAR Section 10.4). Contributions from both direct and air scatter radiation are included in the SKYSHINE-III dose rate calculations. The performance of the SKYSHINE-III code is benchmarked by modeling a set of Kansas State University ^{60}Co skyshine experiments and by modeling two Kansas State University neutron computational benchmarks.

5.4 Shielding Analyses

The shielding evaluations for the NAC-MPC transfer and storage casks are presented in Section 5.4 of the SAR. Shielding calculations are performed using the design basis Yankee Class fuel source terms for fuel with a burnup of 36,000 MWD/MTU and 8 years cooling time. ANSI/ANS Standard 6.1.1-1977 flux-to-dose conversion factors were used to calculate dose rates in the shielding analyses.

5.4.1 Storage Cask

Section 5.4.3 of the SAR provides the dose rate profiles for the NAC-MPC storage and transfer casks based on the design basis fuel source terms. Figures 5.3-1 and 5.3-2 show the NAC-MPC storage cask three-dimensional top and bottom model, respectively. The NAC-MPC storage cask three-dimensional model dose rates are presented in Figures 5.4-1 through 5.4-7 of the SAR. In the vertical dose profile, peaking is observed at the upper and lower end fitting locations, as well as at the locations of the lower inlets and upper outlets. The average and maximum side surface dose rates for the storage cask are 37 mrem/hr and 47.3 mrem/hr, respectively. A maximum dose rate at the surface of the outlets was calculated to be 24

mrem/hr. The maximum dose rate at the entrance of the inlets was calculated to be 99 mrem/hr.

Dose rates were calculated at the top surface of the storage cask and at distances of 1 foot and 1 meter above the lid. Two dose rate peaks were observed on the storage cask top surface. One 90 to 100 cm radially out from the centerline of the storage cask lid, corresponds to the location of the heat transfer annular gap. The other is at approximately 130 cm from the centerline of the lid caused by gammas from the end fitting, top fuel, and top plenum source regions. The average dose rate over the top of the storage cask is calculated to be 25.1 mrem/hr and the peak dose rate is 54 mrem/hr.

5.4.2 Transfer Cask

The transfer cask three-dimensional model dose rates are presented in Figures 5.4-8 through 5.4-15. Figures 5.3-3 and 5.3-4 show the NAC-MPC transfer cask three-dimensional top and bottom model, respectively. The transfer cask dose rate profiles with a wet cavity assumes that the water level in the canister is lowered for welding operations and the fuel's top end fittings are uncovered which causes a large peak in dose rate at the top of the transfer cask. The large peak is due to the gamma contribution from the activated top end fittings. The peak dose rate in this condition is 210 mrem/hr with an average dose rate of 79.5 mrem/hr on the sides of the transfer cask. The peak and average dose rates at 1 meter are 40.5 mrem/hr and 26.4 mrem/hr, respectively. The peak and average dose rates on the bottom of the transfer cask are 77 mrem/hr and 56 mrem/hr, respectively. At 1 meter from the bottom, the peak and average dose rates are 19 mrem/hr and 12 mrem/hr, respectively.

The peak and average dose rates calculated for the transfer cask with a dry cavity are 413 mrem/hr and 226 mrem/hr, respectively. The peak and average dose rates at 1 meter are 103 mrem/hr and 72.2 mrem/hr, respectively. The majority of the dose rate is from the fuel neutron and gamma source but there is a significant contribution from the activated end fittings.

The peak and average transfer cask dose rates with a dry cavity, the shield lid and structural lid in place, and 12.7 cm of temporary steel shielding in place are 359 mrem/hr and 224.6 mrem/hr, respectively. The majority of the dose rate is from the neutrons from the fuel. There are dose rate peaks at the lid edge due to gamma streaming around the edge of the temporary shield. The peak and average dose rates of the bottom of the transfer cask with a dry cavity are 398 mrem/hr and 195 mrem/hr, respectively. At 1 meter from the bottom, the peak and average dose rates are 67 mrem/hr and 28 mrem/hr, respectively.

5.4.3 Off-site Dose Calculations

SKYSHINE-III, Version 5.0.0 was used to calculate off-site dose rates and to determine the minimum distance necessary to achieve a dose rate at the controlled area boundary of 25 mrem/yr as required by 10 CFR 72.104(a). A 2 x 8 cask array was modeled in the code with the source term from each cask represented as top and side surface sources. Using design basis Yankee Class fuel, the SAS1 shielding evaluation provided the surface source emission fluxes. Figure 10.3-1 shows the typical ISFSI 16 cask array layout. Table 10.4-1contains the summary of annual doses at specific distances away from the cask array.

Section 10 of the SER evaluates the overall off-site dose rates from the NAC-MPC. The staff reviewed the SKYSHINE-III evaluation submitted by NAC. The evaluation included the assumptions and design input, cask geometrical model, source spectra, cask array geometrical model and detector locations. NAC also submitted a proprietary version of the SKYSHINE-III NAC Input Manual for familiarization and clarification of the codes application. The staff has reasonable assurance that compliance with 10 CFR 72.104(a) can be achieved by any site licensee. A general licensee who intends to use the NAC-MPC must perform a site-specific evaluation, as required by 10 CFR 72.212(b) demonstrating compliance with 10 CFR 72.104(a). The limit of 25 mrem/year cited in 10 CFR 72.104(a) shall include all site sources. The actual doses to individuals beyond the controlled area boundary depend on site-specific conditions such as cask-array configuration, topography, demographics, and use of engineered features.

Consequently, the final determination of compliance with 72.104(a) is the responsibility of each user under a site license.

5.5 Evaluation Findings

F5.1 The SAR sufficiently describes shielding design features and design criteria for the SSCs important to safety.

F5.2 Radiation shielding features are sufficient to meet the radiation protection requirements of 10 CFR Part 20, 10 CFR 72.104 and 10 CFR 72.106.

F5.3 Operational restrictions to meet dose and as low as is reasonably achievable (ALARA) requirements in 10 CFR Part 20, 10 CFR 72.104, and 10 CFR 72.06 are the responsibility of the site licensee. The NAC-MPC shielding features are designed to assist in meeting these requirements.

F5.4 The design of the shielding system for the NAC-MPC is in compliance with 10 CFR Part 72 and the applicable design and acceptance criteria have been satisfied. The evaluation of the shielding system provides reasonable assurance that the NAC-MPC will provide safe storage of spent fuel. This finding is based on a review that considered the regulations, appropriate regulatory guides, applicable codes and standards, and accepted engineering practices.

6.0 CRITICALITY EVALUATION

The staff's objective in reviewing the applicant's criticality evaluation of the NAC-MPC is to verify that the spent fuel contents remain subcritical under the normal, off-normal, and accident conditions of handling, packaging, transfer, and storage.

The staff reviewed the information provided in Revision 5 of the NAC-MPC SAR to determine whether the NAC-MPC system fulfills the following acceptance criteria, as stated in NUREG-1536:

1. The multiplication factor (k_{eff}), including all biases and uncertainties at a 95% confidence level, should not exceed 0.95 under all credible normal, off-normal, and accident conditions.

2. At least two unlikely, independent, and concurrent or sequential changes to the conditions essential to criticality safety, under normal, off-normal, and accident conditions, should occur before an accidental criticality is deemed to be possible.

3. When practicable, criticality safety of the design should be established on the basis of favorable geometry, permanent fixed neutron-absorbing materials (poisons), or both. Where solid neutron-absorbing materials are used, the design should provide for a positive means to verify their continued efficacy during the storage period.

4. Criticality safety of the cask system should not rely on use of the following credits:

    a. burnup of the fuel
    b. fuel-related burnable neutron absorbers
    c. more than 75% for fixed neutron absorbers when subject to standard acceptance tests.

Observations and conclusions from the staff's review are summarized below.

6.1 Criticality Design Criteria and Features

Criticality safety of the NAC-MPC system depends on the geometry of the fuel basket and the use of fixed Boral panels for absorbing neutrons. The fuel basket features 36 square fuel tubes, each with Boral panels fixed to the four outer walls. The primary design parameters that ensure subcriticality are the minimum flux trap width between fuel tubes and the minimum ^{10}B content of the Boral panels. Each Boral panel in the NAC-MPC has a minimum ^{10}B content of 0.01 g/cm^2. Nominal flux trap widths at various locations in the NAC-MPC basket are 2.58 and 2.33 cm. With neighboring tubes and disk holes fabricated at their combined tolerance extremes, lateral shifting of the tubes allows the widths of individual flux traps to deviate from nominal by up to +0.69 cm or -0.54 cm.

The most reactive credible configurations of the NAC-MPC system occur when the cask is flooded with water. The NAC-MPC does not rely on borated water as a means of criticality control. Therefore, the NAC-MPC would remain subcritical when flooded with pure water. Special features of the basket design, as described in SAR Section 1.2.1.1 and shown in the respective license drawings of SAR Section 1.5.1, ensure the free flow of water between the

fuel tube contents and surrounding flux trap and disk regions of the basket. Uneven flooding within the TSC is therefore not a concern.

The staff reviewed Sections 1, 2, and 6 of Revision 5 of the NAC-MPC SAR and verified that (1) the design features important to criticality safety are clearly identified and adequately described, (2) all criticality-related information shared between Sections 1, 2, and 6 is free of inconsistencies, and (3) the SAR's engineering drawings, figures, and tables are sufficiently detailed to support in-depth review and confirmatory analysis by the staff.

The staff also verified that the design basis off-normal and postulated accident events would not adversely affect the design features important to criticality safety. In terms of maximum system reactivity, the flooded normal configurations, when modeled with waterlogged fuel rods, are identical to or bound the credible configurations resulting from an off-normal or accident event. Based on the information provided in the SAR, the staff concludes that the design of the NAC-MPC system meets the "double contingency" requirements of 10 CFR 72.124(a).

6.2 Fuel Specification

The NAC-MPC system can transfer and store up to 36 intact Yankee Class fuel assemblies. Five vendor categories of Yankee Class assemblies, each with two types of fuel rod configurations, constitute the allowed contents of intact fuel.

Table 6.2-1 of the SAR lists criticality-related parameters for each of the Yankee Class assembly designs approved for transfer and storage in the NAC-MPC system. In that table, the limiting fuel parameters are:

- maximum initial enrichment
- cladding material (Zircaloy or SS348)
- number of fuel rods
- fuel rod pitch
- maximum pellet diameter
- maximum active fuel length
- minimum outer diameter and thickness of cladding
- minimum number and size of displacement rods or guide bars
- displacement-rod or guide-bar material (Zircaloy or SS304L)
- minimum outer diameter and thickness of the instrument tube

In addition, the staff notes that the minimum overall assembly length and minimum length of the bottom fuel hardware are important parameters for restricting the axial location of the active fuel within the TSC basket and, thereby, ensuring its axial coverage by the Boral poison panels under normal and accident conditions. Therefore, the TS also includes lower limits on the overall assembly length and the length of the bottom fuel hardware.

For criticality safety, the most important Yankee Class fuel specifications are the maximum initial fuel enrichment and the fuel cladding material. The NAC-MPC may contain Yankee Class fuel assemblies with maximum initial enrichments of 3.70, 3.90, 4.00, or 4.94 wt% ^{235}U, depending on the vendor category. Only the Westinghouse fuel category with SS348 cladding has an initial enrichment of 4.94 wt% ^{235}U. Because SS348 absorbs more neutrons than the

Zircaloy cladding materials, the 4.94 wt% enriched Westinghouse fuel is less reactive under flooded TSC conditions than the remaining Zircaloy clad Yankee Class fuel types with maximum initial enrichments of 3.70, 3.90, or 4.00 wt% ^{235}U. The most reactive assembly type is the United Nuclear Type A with 4.00 wt% ^{235}U initial enrichment.

Specifications on the condition of the fuel are also included in the SAR and TS. The NAC-MPC system is designed to accommodate intact fuel assemblies, damaged fuel, and fuel debris as defined in the TS. Damaged fuel and fuel debris from the approved Yankee Class assembly types must be placed in RFAs, which are designed to confine fuel pellets and gross fuel particulates to a known, subcritical geometry. With the same outer dimensions as intact fuel assemblies, but with more steel and less than one-third the heavy metal content, RFAs are shown in the SAR to be much less reactive than intact assemblies. One or more of the NAC-MPC's 36 fuel positions may, therefore, contain an RFA in place of an intact assembly.

The staff reviewed the fuel specifications considered in the criticality analyses and verified that they are consistent with or bound the specifications given in Sections 1, 2, and 12 of the SAR and the TS. All fuel assembly parameters important to criticality safety have been included in the TS. The staff confirms that the Yankee Class fuel assembly parameters discussed above bound the maximum reactivity of an NAC-MPC system that does not rely on borated water for criticality control.

6.3 Model Specification

6.3.1 Configuration

The applicant's criticality calculations with KENO-Va apply periodic axial boundary conditions to explicit three-dimensional models of only the central axial region of the fuel and NAC-MPC system. By thus modeling infinitely long fuel assemblies within infinitely long packages, the applicant's analyses neglect the axial reflection of neutrons from one tube to another under full or partial flooding conditions. The staff's independent calculations with MONK8a, however, confirm the applicant's contention that a flooded infinite-length model of the NAC-MPC is never less reactive than the actual-length model with worst-case axial effects (e.g., with partial flooding up to the top of the active fuel).

Sketches of the applicant's calculational models appear in SAR Section 6.3. The models are based on the engineering drawings in SAR Section 1.5 and take into consideration the worst-case dimensional tolerance values. As previously stated, the design basis off-normal and accident events do not affect the performance of the cask design from a criticality standpoint. Therefore, the calculational models for the normal, off-normal, and accident conditions are the same.

To determine the most reactive basket configurations, considering basket component tolerances and relative shifting of the components and fuel, the applicant selectively applied both periodic and mirror boundary conditions to the sides of a KENO-Va model of a single fuel assembly in a basket cell. Results from geometric variations within this basket-lattice model and supplemental variations on a cask model, showed that the most reactive basket configuration for the NAC-MPC transfer and storage casks is with fuel assemblies and fuel tubes shifted toward the center of the package, maximum width of fuel tube openings,

minimum width of disk openings, maximum disk thickness, and minimum spacing of disk openings.

The calculational models also conservatively assumed the following:

- fresh fuel isotopics (i.e., no burnup credit)
- 75% credit for the ^{10}B loading in the Boral panels
- absence of the neutron shield
- flooding of the fuel rod gap regions with pure water.

Intact fuel assemblies were modeled explicitly. To bound the most reactive configuration of fuel debris and damaged fuel, the applicant's model uniformly smeared the maximum allowed mass of the most reactive fuel material within each of the RFA's 64 tubes.

Various moderating conditions, including flooding with full-density and reduced density water, were also considered in the calculational models. As previously stated, the basket is designed to preclude uneven flooding within the TSC.

The staff reviewed the applicant's models and agrees that they are consistent with the design descriptions in SAR Sections 1 and 2, including engineering drawings. The staff also reviewed the applicant's methods, calculations, and results for determining the worst-case manufacturing tolerance. Based on the information presented, the staff agrees that the calculations incorporate the most reactive combination of package parameters and dimensional tolerances.

For its confirmatory analyses with MONK8a, the staff independently modeled the transfer cask using the engineering drawings presented in SAR Section 1.5. Specifically, the staff used Drawing Numbers 455-860, 455-872, 455-881, 455-893, 455-894, and 455-895. Materials used in the staff's analyses were based on MONK8a standard mixtures that closely matched the cask materials indicated in the SAR. The staff's fuel assembly models were derived from fuel design information presented in submitted correspondence from Yankee Atomic Electric Company to the applicant (YRP 372/96). The staff found its models of the cask and contents to be compatible with those of the applicant.

6.3.2 Material Properties

The composition and densities of the materials considered in the calculational models are listed in SAR Sections 6.3.2, 6.3.2.1, 6.3.2.2, and 6.3.2.3.

One of the most important materials in the NAC-MPC is the Boral neutron absorber. The minimum required ^{10}B content is verified through the acceptance testing described in SAR Section 9.1.6. As previously stated, only 75% credit is taken for the ^{10}B content in the Boral panels.

The continued efficacy of the Boral over a 20-year storage period, is assured by the design of the NAC-MPC system. The staff has confirmed that the neutron fluence from irradiated fuel results in negligible depletion of Boral's ^{10}B content and concurs that the Boral panels will remain in place under accident conditions.

The staff reviewed the composition and number densities presented in the SAR and found them to be reasonable. The staff notes that these materials are not unique and are commonly used in other spent fuel storage and transportation applications.

Based on the information provided on the Boral material, the staff agrees that the continued efficacy of the Boral poison can be assured by the design of the NAC-MPC system, and a surveillance or monitoring program is not necessary.

The staff reviewed the neutron absorber acceptance testing described in SAR Section 9.1.6. The staff's acceptance of the absorber testing described in this section is based in part on the fact that the applicant assumed only 75% of the minimum required ^{10}B content in its homogenized model of the Boral poison material.

6.4 Criticality Analysis

6.4.1 Computer Programs

The applicant's principal criticality computational tool was the CSAS25 sequence of SCALE4.3, which invokes the KENO-Va multigroup Monte Carlo code. The calculations used the SCALE system's 27GROUPNDF4, a 27-group cross-section library based on evaluated nuclear data from ENDF/B-IV. Within the CSAS25 sequence, the BONAMI and NITAWL modules of SCALE4.3 provide the necessary problem-specific preprocessing of the 27-group library's resonance data.

The staff agrees that the code and cross-section set used by the applicant are appropriate for this particular application and fuel system. The staff performed its independent criticality analyses using the MONK8a code, a continuous-energy Monte Carlo code with a quasi-pointwise (13,193 energy-group) cross-section library based on evaluated nuclear data from JEF2.2.

6.4.2 Multiplication Factor

The applicant's criticality analyses show that the k_{eff} in the NAC-MPC will remain well below 0.95 for all fuel loadings and conditions. Results of the applicant's CSAS25/KENO-Va criticality calculations for the bounding assemblies are given in Section 6.1 and in Tables 6.4-5 and 6.4-6 of the SAR. The maximum calculated k_{eff} is 0.9021. The staff reviewed the applicant's calculated k_{eff} values and agrees that they have been appropriately adjusted to include all biases and uncertainties at a 95% confidence level or better.

The staff performed independent MONK8a criticality calculations for fully loaded packages under full and partial flooding conditions. Results of the staff's confirmatory calculations were in close agreement with the applicant's results for all five vendor categories of intact Yankee Class fuel.

Based on the applicant's criticality evaluation, as confirmed by the staff, the staff concludes that the NAC-MPC will remain subcritical, with an adequate safety margin, under all credible normal, off-normal, and accident conditions.

6.4.3 Benchmark Comparisons

The applicant performed benchmark calculations on 63 selected critical experiments, chosen, as much as possible, to bound the range of parameters in the NAC-MPC design. The three most important parameters are the ^{10}B loading of the neutron absorbers, the flux trap size, and the fuel enrichment. Parameters such as reflector material and spacing, fuel pellet diameter, and fuel rod pitch were also considered in selecting the critical experiments.

Results of the benchmark calculations show no significant trends in the bias. The benchmark analysis yielded an eigenvalue calculational bias of 0.0052 ± 0.0043. The uncertainty associated with each bias has been multiplied by the one-sided K-factor for 95% probability at the 95% confidence level. The applicant stated that the benchmark and cask calculations were performed with the same computer codes, cross-section data, and computer hardware.

The staff reviewed the applicant's benchmark analysis and agrees that the critical experiments chosen are relevant to the cask design. The staff found the applicant's method for determining the calculational bias acceptable and conservative. The staff also verified that only biases that increase k_{eff} have been applied.

6.5 Supplemental Information

All supportive information has been provided in Revision 5 of the SAR, primarily in Sections 1, 2, 6, and 12.

6.6 Evaluation Findings

Based on the information provided in Revision 5 of the SAR and the staff's own confirmatory analyses, the staff concludes that the NAC-MPC system meets the acceptance criteria specified in NUREG-1536. In addition, the staff finds the following:

F6.1 SSCs important to criticality safety are described in sufficient detail in Sections 1, 2, and 6 of the SAR to enable an evaluation of their effectiveness.

F6.2 The NAC-MPC system is designed to be subcritical under all credible conditions.

F6.3 The criticality design is based on favorable geometry and fixed neutron poisons. An appraisal of the fixed neutron poisons has shown that they will remain effective for the 20-year storage period. In addition, there is no credible way to lose the fixed neutron poisons.

F6.4 The analysis and evaluation of the criticality design and performance have demonstrated that the cask will enable the storage of spent fuel for 20 years with an adequate margin of safety.

F6.5 The staff concludes that the criticality design features for the NAC-MPC system are in compliance with 10 CFR Part 72 and that the applicable design and acceptance criteria have been satisfied. The evaluation of the criticality design provides reasonable assurance that the NAC-MPC system will allow safe storage of spent fuel. In reaching this conclusion, the staff has considered the regulation itself, appropriate regulatory guides, applicable codes and standards, and accepted engineering practices.

7.0 CONFINEMENT EVALUATION

The confinement review ensures that radiological releases to the environment will be within the limits established by the regulations and that the spent fuel cladding and fuel assemblies will be sufficiently protected during storage against degradation that might otherwise lead to gross ruptures. The staff reviewed the information provided in the SAR to determine whether the NAC-MPC system fulfills the following acceptance criteria:

- The SAR must describe the confinement SSCs important to safety in sufficient detail to facilitate evaluation of their effectiveness. [10 CFR 72.24(c)(3) and 10 CFR 72.24(l)]

- The design must adequately protect the spent fuel cladding against degradation that might otherwise lead to gross ruptures during storage, or the fuel must be confined through other means such that fuel degradation during storage will not pose operational safety problems with respect to removal of the fuel from storage. [10 CFR 72.122(h)(1)]

- The cask design must provide redundant sealing of the confinement boundary. [10 CFR 72.236(e)]

- Storage confinement systems must allow continuous monitoring, such that the licensee will be able to determine when to take corrective action to maintain safe storage conditions. [10 CFR 72.122(h)(4) and 10 CFR 72.128(a)(1)]

- The design must provide instrumentation and controls to monitor systems that are important to safety over anticipated ranges for normal and off-normal operations. In addition, the applicant must identify those control systems that must remain operational under accident conditions. [10 CFR 72.122(i)]

- The applicant must estimate the quantity of radionuclides expected to be released annually to the environment. [10 CFR 72.24(l)(1)]

- The applicant must evaluate the cask and its systems important to safety, using appropriate tests or other means acceptable to the Commission, to demonstrate that they will reasonably maintain confinement of radioactive material under normal, off-normal, and credible accident conditions. [10 CFR 72.24(d)]

- SSCs important to safety must be designed to withstand the effects of credible accidents and severe natural phenomena without impairing their capability to perform safety functions. [10 CFR 72.122(b)]

- During normal operations and anticipated occurrences, the annual dose equivalent to any real individual who is located beyond the controlled area must not exceed 25 mrem to the whole body, 75 mrem to the thyroid, and 25 mrem to any other organ. [10 CFR 72.104(a)]

- From any design basis accident, an individual at or beyond the controlled area boundary may not receive the more limiting of (1) a total effective dose equivalent must not exceed 5 rem, or (2) the sum of the deep-dose equivalent plus the committed dose equivalent to any organ may not exceed 50 rem. Additionally, the shallow dose equivalent to the skin or any

extremity shall not exceed 50 rem, and the lens dose equivalent shall not exceed 15 rem. [10 CFR 72.106(a)]

7.1 Confinement Design Characteristics

The staff reviewed the applicant's confinement analyses in SAR Section 7 and the drawings in SAR Section 1. The applicant has clearly identified the confinement boundary. The confinement boundary includes the TSC shell, bottom baseplate, shield lid (including the vent and drain port cover plates), and the associated welds. There are no bolted closures or mechanical seals in the primary confinement boundary. The TSC is designed, fabricated, and tested in accordance with the applicable requirements of the ASME Code, Section III, Subsection NB, to the maximum extent practicable. Exceptions to the ASME Code, with respect to the confinement boundary, are identified in Table 12A4-1 of the SAR. The shield lid (with the vent and drain port cover plates welded to the lid) and the structural lid are independently welded to the upper part of the TSC shell. This design provides redundant sealing of the confinement boundary and satisfies the requirements of 10 CFR 72.236(e). The design, testing, inspection, and examination of the welds forming the confinement boundary are described in detail in Section 7.1.3 of the SAR.

The staff reviewed the cask vacuum drying and backfilling procedures that are used during loading operations. The procedures require that a vacuum pressure of 3 mm of mercury be maintained for 30 minutes without the aid of vacuum equipment to ensure that an acceptably low amount of water and potentially oxidizing material remain in the TSC. The combination of the all-welded cask design and the use of these procedures will ensure that both the cladding the confinement boundary integrity are maintained during normal, off-normal, and hypothetical accident conditions.

The staff also reviewed the applicant's helium leak testing procedures. A leak test of the shield lid is performed to an "as tested leakage rate" of 8×10^{-8} cm^3/sec (helium), e.g., the leaktight standard of 1×10^{-7} std cm^3/sec. The tests will be performed in accordance with ANSI N14.5-1997 using a helium leak detector having a sensitivity of 4×10^{-8} cm^3/sec (helium). During the test, the TSC cavity will be pressurized with helium to 22 psia through the vent port quick disconnect valve. All fittings and connectors used to attach the helium leak testing equipment will be tested to ensure that (1) there is minimal leakage from these sources, and (2) the shield lid-to-shell weld can be tested to leaktight criteria of ANSI N14.5-1997. Any indication of a leak is unacceptable and repair of the leak will be done in accordance with ASME Code Section III. The helium leak test also confirms that the amount of helium lost from the TSC over the license period, at a leakage rate of 8×10^{-8} cm^3/sec (helium), would be less than 1% of the initial amount of helium. Thus, there will be an adequate amount of helium in the TSC to maintain an inert atmosphere and the heat removal capability over the lifetime of the cask.

For normal conditions of storage, the TSC relies on the fuel cladding and the TSC shell cavity as multiple confinement barriers to assure that there is no release of radioactive material to the environment. The TSC is backfilled with an inert gas (helium) to protect against degradation of the cladding. As discussed in Sections 3 and 11 of this SER, there is reasonable assurance that the confinement boundary maintains its structural integrity during normal, off-normal, and hypothetical accident storage conditions. Further, Section 4 of this SER shows that the peak

confinement boundary component temperatures and pressures are within the design basis limits for normal conditions of storage. The integrity of the TSC confinement boundary is assured through (1) nondestructive examinations (NDE), including multiple surface and/or volumetric examinations, of the TSC shield lid, structural lid, and vent and drain port cover plate welds; (2) leakage rate testing; and (3) pneumatic testing. The TSC inspection and test acceptance criteria are described in Section 9.1 of the SAR. TSC closure weld examination and acceptance criteria are described in detail in Section 9.1.1.

The staff concludes that the all-welded construction of the TSC with redundant welded shield and structural lids and associated inspection and testing programs ensure that no release of radioactive material will occur under normal, off-normal, and hypothetical accident conditions.

7.2 Confinement Monitoring Capability

For cask systems having canisters with seal weld closures, continuous monitoring of the weld closures is unnecessary because there is no known plausible, long-term degradation mechanism which would cause the seal welds to fail. However, other licensee monitoring programs, including periodic surveillance, inspection, and radiological and environmental surveys, will ensure that the operating controls and limits are met to maintain safe storage conditions.

7.3 Nuclides with Potential for Release

The confinement boundary of the TSC is designed to be leaktight (i.e., maximum allowable leakage rate of $1x10^{-7}$ std-cm^3/sec) in accordance with ANSI N14.5-1997. In this consensus standard, the definition of leaktight (e.g., "a degree of package containment that, in a practical sense, precludes any significant release of radioactive materials") precludes the need for the applicant to determine the releaseable radiological source term and the corresponding dose consequence. Therefore, the staff concludes that it was unnecessary for the applicant to specify the source term for the confinement analyses.

7.4 Confinement Analysis

The confinement boundary is completely welded, and the stresses, temperatures, and pressures of the TSC are within the design basis limits under normal, off-normal, and hypothetical accident conditions. The TSC is vacuum dried and backfilled with helium gas prior to final canister closure, so there is no potential for an increase in the canister pressure or degradation of the cladding due to radiolytic decomposition or other adverse reactions.

The staff concludes that (1) no discernable leakage of radioactive material from the TSC is credible, (2) the dose consequence due to leakage of radioactive material from the all-welded canister is negligible, and (3) the requirements of 10 CFR 72.104(a) and 10 CFR 72.106(a) are met.

7.5 Supplemental Information

Supplemental information, or documentation, in the form of justifications of assumptions and analytical procedures were provided as requested to complete this review.

7.6 Evaluation Findings

F7.1 Section 7 of the SAR describes confinement structures, systems, and components important to safety in sufficient detail to permit evaluation of their effectiveness.

F7.2 The design of the NAC-MPC adequately protects the spent fuel cladding against degradation that might otherwise lead to gross rupture. Section 4 of the SER discusses the staff's relevant temperature considerations.

F7.3 The design of the NAC-MPC provides redundant sealing of the confinement system closure joints using dual welds on the TSC shield and structural lids.

F7.4 The TSC has no bolted closures or mechanical seals. The confinement boundary contains no external penetrations for pressure monitoring or overpressure protection. No instrumentation is required to remain operational under accident conditions. Since the TSC uses an entirely welded redundant closure system, no direct monitoring of the closures is required.

F7.5 The quantity of radioactive nuclides postulated to be released to the environment is negligible because the TSC is tested to the leaktight standards of ANSI N14.5-1997. The staff concludes that the confinement system will reasonably maintain confinement of radioactive material under normal, off-normal, and credible accident conditions. The corresponding dose from leakage of radioactive material from the TSC is also negligible. Thus, the NAC-MPC satisfies the regulatory requirements of 10 CFR 72.104(a) and 10 CFR 72.106(b).

F7.6 The staff concludes that the design of the confinement system of the NAC-MPC is in compliance with 10 CFR Part 72 and that the applicable design and acceptance criteria have been satisfied. The evaluation of the confinement system design provides reasonable assurance that the NAC-MPC will allow safe storage of spent fuel. This finding is based on a review of the regulation itself, appropriate regulatory guides, applicable codes and standards, the applicant's analysis, the staff's confirmatory analysis, and accepted engineering practices.

8.0 OPERATING PROCEDURES

The staff reviews the operating procedures descriptions to ensure that the applicant's SAR presents acceptable operating sequences, guidance, and generic procedures for cask loading, handling/storage, and unloading operations. The procedures incorporate and are compatible with the applicable operating and control limits specified in the TS. The operating procedures properly consider the prevention of hydrogen gas generation from any cause and include appropriate precautions to minimize occupational radiation exposures ALARA. The operating procedures contain a table listing the ancillary equipment necessary to support ISFSI loading, storage, and unloading operations. Detailed procedures, including prerequisites, preparation, receipt inspection, loading, handling, and unloading procedures must be developed by each cask user in accordance with an approved QA program.

8.1 Cask Loading

The cask loading operations described in the SAR include the appropriate provisions to be accomplished before cask loading. These include a visual inspection of the basket fuel tubes for obstructions, verification of shielding levels, temperature requirements for lifting of the transfer cask and the use of an approved loading path for the heavy lift. The loading procedure describes the activities sequentially in the anticipated order of performance.

8.1.1 Fuel Specifications

The procedures described in the SAR provide for fuel assembly selection verification by the user to verify conformance with the detailed authorized contents listed in the TS. Precise fuel specifications for fuel that is permitted to be loaded into a TSC are specifically designated in the TS. Detailed site-specific procedures are necessary to ensure all fuel loaded in the cask meets the fuel specifications as delineated in the CoC and the TS. These procedures are subject to evaluation on a site-specific basis through the inspection process rather than during the licensing review.

8.1.2 ALARA

The NAC-MPC cask loading procedures incorporate general ALARA principles and practices. ALARA practices include the use of temporary shielding during the setup of the automatic welding equipment, the use of automated welding equipment, and performing certain operations (decontamination of the exterior surface of the transfer cask, welding of the shield lid, and pressure testing of the TSC) while the TSC remains filled with water. The procedures incorporate TS 3.2.2, which specifies limits for removable contamination on accessible surfaces of the TSC and transfer cask. Each cask user will need to develop detailed loading procedures that incorporate the ALARA objectives of their site-specific radiation protection program.

8.1.3 Draining and Drying

The operating procedures for draining the water from and vacuum drying the TSC can be found in SAR Section 8.1. These procedures clearly describe the process of removing water vapor and oxidizing material to acceptable levels from the cask.

Once the shield lid has been welded in place and the PT examination of the weld has been completed, a suction pump is attached to the drain line, and the water in the cask is removed while the hose connected to the vent port remains open. After the drain port cover is welded to the shield lid and nondestructively examined, a vacuum system is connected to the vent port hose. The vacuum system is used to evacuate the air and water vapor from the cask until a steady pressure of less than or equal to 3 millimeters of mercury (mm Hg) is achieved, with the pump isolated, for 30 minutes. Then, the cask is backfilled with helium gas before a second cycle of vacuum drying (3 mm Hg for 30 minutes) is performed. Finally, the cask cavity is backfilled with helium (99.9% minimum purity) to 22 psia for subsequent helium leak testing. The operating controls and limits for this procedure are described in more detail in Section 12 of the SAR.

The vacuum pressure of 3 mm Hg prescribed for the vacuum drying procedure is consistent with methodology described in NUREG-1536, which references PNL-6365. Moisture removal is inherent in the vacuum drying process, and levels at or below those evaluated in PNL-6365 are expected if the vacuum drying is performed as described in the SAR. This procedure will serve to reduce the amount of oxidants to below the levels where significant cladding degradation is expected.

The staff concludes that: (1) helium (99.9% minimum purity) is an acceptable inert cover gas to minimize the source of potentially oxidizing impurity gases and vapors and (2) the NAC-MPC operating procedures (i.e., two cycles of alternating vacuum drying and backfilling with a high purity cover gas) are adequate to sufficiently remove contaminants from the cask.

8.1.4 Welding and Sealing

A general description of the Automated Welding System can be found in SAR Section 1.2.1.4. This system will be used to weld the inner and outer lid closure welds of the TSC during cask loading operations to ensure the dose to welders will be ALARA. Prior to welding the shield lid (i.e., the inner closure weld), approximately 50 gallons of water will be removed from the TSC to keep moisture away from the weld region. Section 8.1 of the SAR describes the loading procedures that incorporate the welding, NDE, helium leak test, and hydrostatic test procedures. As indicated in SAR Section 7.1.3, unacceptable weld defects will be repaired in accordance with ASME Code Section III, Subarticle NB-4450, and re-examined. The staff concludes the procedures for welding and NDE of the closure welds are acceptable.

Leak testing will be performed to demonstrate the leaktightness of the TSC shield lid in accordance with ANSI N14.5-1997. The applicant states that the test will demonstrate a leak rate less than 8.0×10^{-8} cm^3/s helium, performed at a minimum sensitivity of 4.0×10^{-8} cm^3/s helium, in accordance with SAR Section 9 and the TS. The staff concludes these procedures provide for acceptable welding and NDE of the closure welds.

SAR Section 8 also describes the welding of the redundant TSC structural lid, which is placed over the shield lid. The structural lid to canister shell weld is either: (1) UT examined, with the final surface PT examined, in accordance with ASME Code Section V, or (2) progressively PT

examined in accordance with ASME Code Section V. SAR Section 8 also describes the installation of the VCC lid, including bolts and tamper indication devices. The appropriate bolt torque values are listed in Table 8.1-2 of the SAR. The staff concludes these procedures provide for acceptable sealing of the TSC structural lid and the VCC lid.

8.2 Cask Handling and Storage Operations

All accident events applicable to the transfer of the TSC to the VCC and of the VCC to the storage location are bounded by the design events described in Sections 2, 5, and 11 of the SAR. All conditions for lifting and handling methods are bounded by the evaluations in Sections 3 and 4 of the SAR. There are TS associated with VCC loading and transfer operations, such as restricting lift heights and environmental conditions.

Inspection, surveillance, and maintenance requirements that are applicable during ISFSI storage are discussed in Section 9 of the SAR. Surveillance and monitoring requirements are included in Section 12 of the SAR. The technical specifications which comprise Section 12 of the SAR have been appended to the CoC. The staff determined that these discussions were acceptable.

Occupational and public exposure estimates are evaluated in Section 10 of the SAR. Each cask user will need to develop detailed cask handling and storage procedures that incorporate the ALARA objectives of their site-specific radiation protection program.

8.3 Cask Unloading

Detailed unloading procedures must be developed by each cask user.

The NAC-MPC system unloading procedures describe the general actions necessary to remove the TSC from the VCC and to unload the TSC. The operating procedure for transferring a loaded TSC from the VCC to the NAC-STC transport cask is discussed in the NAC-STC SAR and is not evaluated in this SER. Special precautions are outlined to ensure personnel safety during the unloading operations.

8.3.1 Cooling Venting & Reflooding

The operating procedures in Section 8 of the SAR specify, prior to initiating cooldown, the sampling for radioactive gases and the subsequent flushing of the radioactive gases with nitrogen while monitoring the exit temperatures. A cooldown system is subsequently attached to the drain connection (inlet) and the vent connection (outlet). A controlled water flow rate, with a specified minimum water temperature is established with the steam and water being discharged to the spent fuel pool or radioactive water treatment system. The applicant's evaluation of the controlled TSC reflooding and cooling of the stored fuel assemblies determined that the associated thermal stresses on cladding and the steam pressures developed within the canister are acceptable. The procedures reflect the appropriate TS which stipulates the minimum cooling water temperature, maximum cooling water flow rate, and maximum canister pressure.

Procedures for obtaining a gas sample are included to provide for assessment of the condition of the fuel assembly cladding. This allows for detection of potentially damaged or oxidized fuel. The procedures include ALARA caution steps to prevent the possible spread of contamination and allow for the implementation of additional measures appropriate for the specific conditions.

8.3.2 ALARA

The NAC-MPC cask unloading procedure descriptions incorporate general ALARA principles and practices. ALARA practices include provisions for radiological surveys, exposure and contamination control measures, temporary shielding, and caution statements related to specific actions that could change radiological conditions. Each cask user will need to develop detailed unloading procedures that incorporate the ALARA objectives of their site-specific radiation protection program.

8.4 Evaluation Findings

F8.1 The NAC-MPC is compatible with wet loading and unloading. General procedure descriptions for these operations are summarized in Sections 8.1 and 8.3 of the SAR. Detailed procedures will need to be developed and approved on a site-specific basis.

F8.2 The bolted VCC closure and welded TSC shield and structural lids of the cask allow ready retrieval of the spent fuel for further processing or disposal as required.

F8.3 The contents of the general operating procedures are designed to minimize and facilitate decontamination. Routine decontamination will be necessary after the cask is removed from the spent fuel pool.

F8.4 No significant radioactive effluents are produced during storage. Any radioactive effluents generated during the cask loading and unloading will be governed by the 10 CFR Part 50 license conditions.

F8.5 The contents of the general operating procedures described in the SAR are adequate to protect health and minimize danger to life and property. Detailed procedures will need to be developed and approved on a site-specific basis.

F8.6 Section 10 of the SER assesses the operational restrictions to meet the limits of 10 CFR Part 20. Additional site-specific restrictions may also be established by the site licensee.

F8.7 The staff concludes that the contents of the generic procedures and guidance for the operation of the NAC-MPC are in compliance with 10 CFR Part 72 and that the applicable acceptance criteria have been satisfied. The evaluation of the operating procedure descriptions provided in the SAR offers reasonable assurance that the cask will enable safe storage of spent fuel. This finding is based on a review that considered the regulations, appropriate regulatory guides, applicable codes and standards, and accepted practices.

9.0 ACCEPTANCE TESTS AND MAINTENANCE PROGRAM

The objective of the review of the acceptance tests and maintenance program is to ensure that the SAR includes the appropriate acceptance tests and maintenance programs for the NAC-MPC. A clear, specific listing of these commitments will help avoid ambiguities concerning design, fabrication, and operational testing requirements when the NRC staff conducts subsequent inspections.

9.1 Acceptance Tests

All materials and components will be procured with certifications and supporting documentation, as necessary to assure compliance with procurement specifications, and are receipt inspected for visual and dimensional acceptability, conformance to materials certifications, and materials traceability. A fit-up test of the canister and its components will be performed during acceptance inspection to ensure that the canister, basket, shield lid, and structural lid can be properly assembled in the field during fuel loading and canister closure.

9.1.1 Visual and Nondestructive Examination Inspections

Except as identified herein, the NAC-MPC confinement boundary is fabricated and inspected in accordance with ASME Code Section III, Subsection NB. Exceptions to the ASME Code are identified in the TS and include: (1) partial penetration welds of the shield lid- and structural lid-to-shell joints, (2) a remaining backing ring that is used to weld the structural lid to the shell, (3) root and final surface PT examination of the shield lid-to-shell weld and the vent and drain port cover-to-shield lid welds, and (4) either UT or progressive PT examination of the structural lid-to-shell weld. The shield and structural lids are welded independently to provide a redundant seal. The staff reviewed these exceptions, and the corresponding justifications, and found them to be in accordance with the guidelines of NUREG-1536 with the following clarifications:

> In the confinement Section of NUREG-1536, it states that the staff has accepted meeting the examination requirements of the ASME Section III Code for Class 1 or Class 2 components. These Code requirements require a volumetric examination of the canister closure weld, however, NRC's ISG No.4 permits the use of multilayer PT surface examination as a substitute for the ASME Code required volumetric examination.
>
> The acceptance criteria for the UT volumetric examination of the closure weld shall be as stated in Paragraph NB-5332 of the ASME Section III Code which allows the use of a Section XI fracture mechanics to justify maximum flaw size, in lieu of, the no crack criteria of Subarticle NB-5330.

The basket, basket support disks, and fuel tubes are fabricated and inspected in accordance with ASME Code, Section III, Subsection NG. The transfer cask is designed and fabricated in accordance with ANSI N14.6 and NUREG-0612. Welding of the VCC steel components is performed in accordance with either AWS D1.1-96, with visual inspection requirements contained in Section 8.15.1, or ASME Code Section VIII, using VT and magnetic particle (MT) examination techniques of ASME Code Section V.

The NDE of weldments is well-characterized on the drawings, and standard NDE symbols and/or notations are used in accordance with AWS 2.4, "Standard Symbols for Welding, Brazing, and Nondestructive Examination." Fabrication inspections include VT, PT, MT, UT, and RT examinations, as applicable.

Structural and confinement boundary weld examinations and acceptance criteria, in general, meet the applicable requirements of ASME Code, Section III. The majority of confinement boundary welds are volumetrically examined in accordance with Code requirements using RT with acceptance criteria per NB-5320. The bottom plate to canister shell weld is volumetrically examined in the shop, using UT, with acceptance criteria per NB-5330. For the confinement boundary welds made in the field, all will have their root and final weld passes PT examined. However, the closure weld for the structural lid-to-canister shell will be either (1) progressively PT examined with each layer not to exceed 0.375 inch or (2) UT examined. Use of a progressive PT examination for the confinement boundary welds is currently not in agreement with ASME Section III, Class 1 requirements. However, it is acceptable per NRC's ISG No.4. The distance between progressive layered PT was justified per a fracture mechanics analysis which calculated a critical flaw size as discussed in Section 3.6 of the SAR. The calculation of the critical flaw size of the closure weld assumes a 360 degree flaw that could exist under the weld pass surface that is PT examined. As allowed by the NRC's ISG No.4, postulated cracks under each PT examined surface are not required to be additive for comparison to the critical flaw size. The staff finds that the closure weld for the structural lid may be inspected using either volumetric or multiple pass dye penetrant techniques subject to the following conditions, as stated in ISG No.4:

1) PT examinations may be used in lieu of volumetric examinations only on austenitic stainless steels. PT should be done in accordance with ASME Section V, Article 6, "Liquid Penetrant Examination."

2) For either UT or multiple layer PT, the minimum detectable flaw size must be demonstrated to be less than the critical flaw size. The critical flaw size shall be calculated in accordance with ASME Section XI methodology; however, net section stress may be governing for austenitic stainless steels and must not violate Section III requirements.

3) If PT alone is used, at a minimum, it must include the root and final layers and sufficient intermediate layers to detect critical flaws.

4) The inspection of the weld must be performed by qualified personnel and shall meet the acceptance requirements of ASME Code Section III, NB-5350, for PT and NB-5332 for UT examinations.

5) If PT alone is used, a design stress-reduction factor of 0.8 must be applied to the weld design.

6) The results of the PT examination, including all relevant indications, shall be made a permanent part of the licensee's records by video, photographic, or other means providing a retrievable record of weld integrity. Video or photographic records should be taken during the final interpretation period described in ASME Section V, Article 6, T-676.

The staff finds that the NDE and acceptance criteria used on the NAC-MPC are acceptable based on meeting the governing Code's requirements or are permitted via guidance provided to the staff (i.e., ISG No.4).

9.1.2 Structural/Pressure Tests

9.1.2.1 Lifting Trunnions

The transfer cask lifting trunnions and bottom shield doors are load tested in accordance with ANSI N14.6. The lifting trunnions are tested by applying a vertical load of 429,039 lbs, which is 300 % of the maximum service load. Similarly, the bottom shield doors are tested by applying a vertical load of 186,810 lbs, which is 300 % of the maximum service load. The loads are held for a minimum of 10 minutes. Following the load tests, all trunnion and door rail welds and all load bearing surfaces are visually inspected for permanent deformation, galling, or cracking and examined using MT or PT methods. The acceptance criteria for the MT and PT examinations are in accordance with ASME Code Section III, NF-5340 and NF-5350, respectively.

9.1.2.2 Pneumatic Pressure Testing

The TSC is pressure tested after the shield lid is welded in place and after approximately 50 gallons of water have been removed from the canister. The pneumatic testing of the canister and shield lid weld is performed at 1.2 times the normal conditions design pressure, in accordance with ASME Code, Section III, Subsection NB. A pressure of 50 psig is held for a minimum of 10 minutes with no loss of pressure. Following completion of the test, a final dye penetrant examination of the shield lid weld is performed.

9.1.2.3 Leak Testing

A helium leak test is performed to verify that the shield lid weld is leaktight as defined by ANSI N14.5-1997. The containment vessel is pressurized with helium to 22 psia. The test demonstrates a leakage rate of less than 8.0×10^{-8} cm^3/sec (helium) and is measured to a sensitivity of at least 4.0×10^{-8} cm^3/sec (helium). Any indication of a helium leak is unacceptable and must be repaired.

9.1.3 Shielding Tests

The storage cask radial shield design consists of a 3.5-inch thick carbon steel inner liner surrounded by 21 inches of reinforced concrete. Gamma shielding is provided by both the carbon steel and concrete, and neutron shielding is provided primarily by the concrete. Additional radial shielding is provided by the TSC stainless steel shell. The storage cask top shielding design is comprised of 8 inches of stainless steel from the canister lids, a shield plug containing 1 inch of NS-4-FR encased within 4.125 inches of carbon steel, and a 1.5-inch thick carbon steel lid. The bottom shielding of the concrete cask consists of the 1 inch stainless steel canister bottom and 3 inches of carbon steel plate. Construction of the NAC-MPC system is in accordance with detailed fabrication specifications, with all fabrication activities performed in accordance with NRC approved QA programs.

The effectiveness of the neutron and gamma shielding for the storage cask is verified by the performance of external dose rate surveys. TS 3.2.1 limits the acceptable storage cask average radiation doses due to gammas and neutrons to 50 mr/hr, 35 mr/hr, and 100 mr/hr for the cask side, top, and inlets and outlets, respectively.

9.1.4 Neutron Absorber Tests

After manufacturing, each lot of Boral is tested using wet chemistry and/or neutron attenuation techniques to verify the minimum ^{10}B content. The test shall be representative of each Boral panel. The minimum allowable ^{10}B content is 0.01 g/cm^2 for the NAC-MPC Boral panels. Any panel with a ^{10}B loading less than the minimum allowed will be rejected.

The staff's acceptance of the neutron absorber test described above is based, in part, on the fact that the criticality analyses assumed only 75% of the minimum required ^{10}B content of the Boral. For greater credit allowance, special, comprehensive fabrication tests capable of verifying the presence, uniformity, and particle-size distribution of the neutron absorber are necessary.

Installation of the Boral panels on the fuel basket tubes shall be performed in accordance with written and approved procedures. Quality control procedures shall be in place to ensure that each required tube wall of the TSC basket contains a Boral panel in accordance with NAC Drawing No. 455-881.

9.1.5 Thermal Tests

A thermal performance program monitors daily, for each NAC-MPC system cask, the outlet temperature indicators. The outlet temperatures are recorded, compared with the ambient air temperature and verified to have less than a 92°F differential.

The first NAC-MPC system in place that has a heat load of greater than 7.5 kW will be analyzed by temperature measurements to confirm the overall heat transfer characteristics of the system. Special Requirement 5.3 of the TS requires the submittal of the results of the temperature measurements for the highest heat load in the NAC-MPC system, up to the authorized maximum of 12.5 kW.

9.1.6 Cask Identification

The TSC will be marked with a model number, unique identification number, and empty weight. This information will appear on a data plate, which is detailed in drawings in SAR Section 1. In addition, the exterior of the VCC which will hold the TSC while it is in storage will be marked. This marking provides a unique, permanent, and visible number to permit identification.

9.2 Maintenance Program

The NAC-MPC storage system is a passive system with a minimum amount of lifetime required maintenance. Each storage cask is visually inspected daily to detect any blockage of air inlets and outlets and to verify the protection screens are in place. The temperatures of the cask outlets are monitored daily. The concrete cask is annually visually inspected for chipping, spalling, or other surface defects.

9.3 Evaluation Findings

F9.1. Section 9.1 of the SAR describes the applicant's proposed program for pre-operational testing and initial operations of the NAC-MPC system. Section 9.2 discusses the proposed maintenance program.

F9.2 SSCs important to safety will be designed, fabricated, erected, tested, and maintained to quality standards commensurate with the importance to safety of the function they are intended to perform. Tables 2.3-1 identifies the safety importance of SSCs, and Section 1.2 of the SAR presents the applicable standards for their design, fabrication, and testing.

F9.3 The certificate holder/licensee will examine and/or test the NAC-MPC to ensure that it does not exhibit any defects that could significantly reduce its confinement effectiveness. Section 9.1 of the SAR describes this inspection and testing.

F9.4 The certificate holder/licensee will mark the TSC with a steel stamp indicating its model number, unique identification number, and empty weight. Drawing 455-871 illustrates and describes this data.

F9.5 The staff concludes that the acceptance tests and maintenance program for the NAC-MPC system are in compliance with 10 CFR Part 72 and that the applicable acceptance criteria have been satisfied. The evaluation of the acceptance tests and maintenance program provides reasonable assurance that the cask will allow safe storage of spent fuel throughout its licensed or certified term. This finding is reached on the basis of a review that considered the regulation itself, appropriate regulatory guides, applicable codes and standards, and accepted practices.

10.0 RADIATION PROTECTION EVALUATION

This section evaluates the capability of the radiation protection design features, design criteria, and operating procedures of the NAC-MPC to meet regulatory dose requirements. The regulatory requirements for providing adequate radiation protection to site licensee personnel and members of the public include 10 CFR Part 20, 10 CFR 72.104(a), 72.106(b), 72.212(b), and 72.236(d).

10.1 Radiation Protection Design Criteria and Design Features

10.1.1 Design Criteria

Section 10.1 of the SAR describes the radiological protection design criteria of the NAC-MPC to meet the limits and requirements in 10 CFR Part 20, 10 CFR 72.104, 10 CFR 72.106, and the guidance in Regulatory Guide 8.8. As required by 10 CFR 72.212, a general licensee will be responsible for demonstrating site-specific compliance with these requirements.

10.1.2 Design Features

A general description of the NAC-MPC system is contained in Section 1 of the SAR. Section 10.1.2 of the SAR presents those design features which provide radiation protection to both on-site workers and members of the public beyond the controlled area. Design features discussed in the SAR include the following:

- Concrete-walled overpack which provides shielding of gamma and neutron radiation.
- A confinement system with multiple welded barriers to prevent atmospheric release of radionuclides.
- Operating procedures which incorporate ALARA principles to reduce occupational exposures.
- Material selection and surface preparation that facilitates decontamination.
- Positive clean water flow in the transfer cask/canister annulus to minimize the potential for contamination during in-pool fuel loading.
- The low-maintenance design reduces occupational doses during storage.
- The confinement system is designed to maintain confinement of fuel during accident conditions.
- Nonplanar cooling air pathways to minimize radiation streaming at the inlets and outlets of the concrete cask.

The staff evaluated the radiation protection design features and design criteria for the NAC-MPC and found them acceptable. The SAR analysis provides reasonable assurance that use of the NAC-MPC storage cask can meet the regulatory requirements in 10 CFR Part 20, 10 CFR 72.104(a), and 10 CFR 72.106(b).

10.2 ALARA

Section 10.1 of the SAR presents the ALARA considerations for the NAC-MPC. Radiation protection design features and the design criteria address ALARA requirements consistent with the requirements in 10 CFR 20 and guidance provided in Regulatory Guides 8.8 and 8.10.

The NAC-MPC design features are designed to maintain radiation exposures ALARA and the TS provide surface dose rates and surface contamination limits for the NAC-MPC. Section 10.1.3 of the SAR includes the operational considerations for ALARA and describes optional auxiliary shielding devices to minimize occupational and public doses. The specific operational considerations to be used by a particular licensee will be determined by the user's operational conditions and facilities.

Each general licensee, in accordance with 10 CFR 72.212, will implement its existing site-specific radiation protection program, ALARA policies, and procedures for cask operations to ensure that personnel exposure requirements in 10 CFR Part 20 are met.

The staff evaluated the ALARA elements incorporated into the NAC-MPC design and found them to be acceptable. Operational ALARA policies, procedures, and practices are the responsibility of the site licensee as required by 10 CFR Part 20. In addition, TS 3.2.1 and 3.2.2 establish limits for the surface dose rates and surface contamination, respectively, for the NAC-MPC.

10.3 Occupational Exposures

Section 8 of the SAR discusses the general operating procedures that licensees will use for fuel loading, cask operation, and fuel unloading. Section 10.3 of the SAR discusses the estimated number of personnel, the estimated dose rates, and the estimated time for each task. The estimated occupational person-rem is based upon the minimum number of personnel needed to accomplish the requirements from the general operating procedures and the dose rates determined from the shielding evaluation in Section 5 of the SAR. The estimated occupational person-rem exposure for loading the cask and transferring to the storage pad is presented in Table 10.3.1 of the SAR. The dose estimates indicate the total occupational dose in loading a single cask with design basis fuel (shielding) for the NAC-MPC is approximately 2.2 person-rem. The estimated occupational exposure for unloading the cask is 2.0 person-rem. The yearly estimated exposure for surveillance and cask maintenance for a 16 cask array is approximately 0.626 person-rem.

The staff reviewed the estimated occupational exposures and found them to be acceptable. The occupational exposure dose estimates provide reasonable assurance that occupational limits in 10 CFR Part 20 Subpart C can be achieved. Actual occupational doses will depend on site-specific parameters, including special measures taken to maintain exposures ALARA. Each licensee will have an established radiation protection program, as required in 10 CFR Part 20 Subpart B. In addition, each licensee must demonstrate compliance with all dose limits in 10 CFR Part 20, 10 CFR Part 72, and any site-specific 10 CFR Part 50 license requirements with evaluations prior to loading of the casks.

10.4 Public Exposures

Section 10.4 of the SAR summarizes the calculated dose rates to individuals beyond the controlled area (members of the public), as determined from Section 5 of the SAR. Based on the containment evaluation of Section 7 of the SAR, no effluents are expected from the storage cask during normal, off-normal, or accident conditions because of the leaktight configuration of the TSC. As determined from the containment evaluation, the confinement

boundary of the TSC is designed to be leaktight, and therefore, no discernable leakage of radioactive material from the TSC is credible. Therefore, direct radiation (including skyshine) is the primary dose pathway to individuals beyond the controlled area during normal and off-normal conditions. A discussion of the staff's evaluation and confirmatory analysis of the shielding and confinement dose calculations are presented in Sections 5 and 7 of the SER.

Public exposure from normal and off-normal conditions will be from direct and reflected radiation from the storage casks. Table 10.4-1 presents a summary of the results of the SKYSHINE-III evaluation to determine the minimum distance necessary to achieve an annual dose of 25 mrem from a 2 by 8 array of casks. Based upon the evaluation, NAC has determined that for a 2 by 8 array of design basis fuel, an enveloping boundary 200 meters by 150 meters around the ISFSI will ensure compliance with the dose limit in 10 CFR 72.104(a). Each licensee who intends to use the NAC-MPC must perform a site-specific dose analysis to demonstrate compliance with all the requirements in 10 CFR 72. Site-specific boundary distances may vary based on fuel type, fuel cooling time, natural site barriers, and number of casks in service.

The staff evaluated the public dose estimates from direct and reflected radiation from normal and off-normal (anticipated occurrences) conditions and found them to be acceptable. The staff has reasonable assurance that compliance with 10 CFR 72.104(a) can be achieved by each site licensee. The general license holder must perform a site-specific evaluation, as required by 10 CFR 72.212(b), to demonstrate compliance with 10 CFR 72.104(a). The actual doses to individuals beyond the controlled area boundary depend on site-specific conditions such as cask array configuration, topography, and use of engineered features (e.g., berms). In addition, the dose limits in 10 CFR 72.104(a) include doses from other fuel cycle activities such as reactor operations. Consequently, final determination of compliance with 10 CFR 72.104(a) is the responsibility of each site licensee.

The licensee will also have an established radiation protection program as required by 10 CFR Part 20, Subpart B, and will demonstrate compliance with dose limits to individual members of the public, as required by 10 CFR Part 20, Subpart D, by evaluations and measurements.

Section 11 of the SAR contains a description of accident conditions and natural phenomena events which could affect the ISFSI. The SAR evaluation concluded that the confinement function of the NAC-MPC is not breached by design basis accidents or natural phenomena events.

The staff evaluated the public dose estimates from direct radiation from accident conditions and natural phenomena events and found them acceptable. A discussion of the staff's evaluation and confirmatory analysis of the shielding and confinement dose calculations are presented in Sections 5 and 7 of the SER, respectively. The staff has reasonable assurance that the effects of direct radiation from bounding design basis accidents and natural phenomena will be below the regulatory limit of 5 rem specified in 10 CFR 72.106(b).

10.5 Evaluation Findings

F10.1 The SAR sufficiently describes radiation protection design bases and design criteria for the SSCs important to safety for the NAC-MPC.

F10.2 Radiation shielding and confinement features are sufficient to meet the radiation protection requirements of 10 CFR Part 20, 10 CFR 72.104, and 10 CFR 72.106.

F10.3 The NAC-MPC is designed to provide redundant sealing of confinement systems.

F10.4 The NAC-MPC is designed to facilitate decontamination to the extent practicable.

F10.5 The SAR adequately evaluates the NAC-MPC and its systems important to safety to demonstrate that they will reasonably maintain confinement of radioactive material under normal, off-normal, and accident conditions.

F10.6 The SAR sufficiently describes the means for controlling and limiting occupational exposures within the dose and ALARA requirements of 10 CFR Part 20.

F10.7 Operational restrictions to meet dose and ALARA requirements in 10 CFR 20, 10 CFR 72.104, and 10 CFR 72.106 are the responsibility of the site licensee. The NAC-MPC is designed to assist in meeting these requirements.

F10.8 The staff concludes that the design of the radiation protection system for the NAC-MPC is in compliance with 10 CFR Part 72 and the applicable design and acceptance criteria have been satisfied. The evaluation of the radiation protection system design provides reasonable assurance that the NAC-MPC will provide safe storage of spent fuel. This finding is based on a review that considered the regulation itself, appropriate regulatory guides, applicable codes and standards, and accepted engineering practices.

11.0 ACCIDENT ANALYSES

The purpose of the review of the accident analyses is to evaluate the applicant's identification and analysis of hazards, as well as the summary analysis of system responses to both off-normal and accident or design basis events. This ensures that the applicant has conducted thorough accident analyses, as reflected by the following factors:

1. identified all credible accidents
2. provided complete information in the SAR
3. analyzed the safety performance of the cask system in each review area
4. fulfilled all applicable regulatory requirements

11.1 Off-Normal Events

Section 11.1 of the SAR examines the causes, radiological consequences, system performance, and corrective actions for off-normal events. The SAR determined that the confinement function of the NAC-MPC is not affected by off-normal conditions.

The staff reviewed the off-normal event analyses with respect to 10 CFR 72.104(a) dose limits and found them acceptable. The staff has reasonable assurance that the dose to any individual beyond the controlled area will not exceed the limits in 10 CFR 72.104(a) during off-normal conditions (anticipated occurrences).

11.1.1 Blockage of Half of the Air Inlets

The applicant evaluated the NAC-MPC for conditions associated with half of the cask inlets blocked, including an environmental temperature of 75°F and a 12-hour insolation period of 2950 BTU/ft^2 and 1475 BTU/ft^2 for horizontal flat and curved surfaces, respectively. The analysis showed that the resultant component temperatures are less than the allowable temperatures. The personnel dose received as a result of clearing the blockage was estimated to be a maximum of 60 mrem to extremities. The staff concludes that the effects and consequences of this off-normal event are in compliance with the radiological dose limits from normal operations and anticipated occurrences provided in 10 CFR 72.104(a).

11.1.2 Canister Off-Normal Handling Load

The applicant evaluated the consequences of loads on the TSC during the installation of the canister in the VCC or removal of the TSC from the VCC or transfer cask. In SER Section 3.3.3, the staff reviewed the SAR evaluation and concurred with its conclusion on the positive structural margin of safety. This demonstrates that the TSC will reasonably maintain confinement of radioactive material under the off-normal handling condition and that there are no radiological consequences for this event.

11.1.3 Failure of Instrumentation

The applicant evaluated the failure of the electronic temperature monitoring instrumentation. Since temperature recordings and surveillance of the cask inlets and outlets occurs daily, the SAR Section 11.2 analysis for complete blockage of the air inlets and outlets is bounding for component temperatures. There are no radiological consequences for this event.

11.1.4 Severe Environmental Conditions (100°F and -40°F)

The applicant evaluated the NAC-MPC for a severe environmental heat of 100°F and a 12-hour insolation period of 2950 BTU/ft^2 and 1475 BTU/ft^2 for horizontal flat and curved surfaces, respectively. Also, the maximum decay heat of 12.5 kW was modeled as identified in Section 11.1.4 of the SAR.

The applicant also evaluated the NAC-MPC for conditions with ambient temperatures of -40°F, no solar insolation, and applied the maximum decay heat of 12.5 kW, as described in Section 11.1.4 of the SAR . The staff concurs with this approach since the largest radial thermal gradient would exist with the maximum decay heat load and, thus, produce the largest thermal stresses. Also, since the material of the canister is a ductile stainless steel, it would not be susceptible to brittle fracture associated with the colder temperatures.

The evaluations show that the component temperatures are within the allowable values for the off-normal ambient conditions. There are no radiological consequences for this event.

11.1.5 Small Release of Radioactive Particulate - Canister Exterior

The applicant evaluated the effects of the airborne release of canister surface contamination as a result of air flow over the canister surface. The applicant calculated the surface contamination level for a 16 cask array which would result in an annual dose of 1 mrem at 100 meters. Such surface contamination levels are at least 10 times higher than the surface contamination limits in the TS for the accessible surfaces of the canister. The calculated low annual dose at 100 meters is a negligible radiological consequence.

11.2 Design Basis Accidents and Natural Phenomena Events

Section 11.2 of the SAR determines the radiological dose consequences for the identified design basis accidents and natural phenomena events. The SAR determined that the NAC-MPC system has adequate design margins and would reasonably maintain its confinement function during and after design basis accidents. As a result, it is reasonable to expect that any radioactive material releases and increased radiological doses from design basis accidents and natural phenomena events would be small. The applicant determined that the radiological dose at 100 meters would not exceed the dose limits specified in 10 CFR 72.106(b).

The staff reviewed the design basis accident analyses, performed confirmatory analyses, and found the NAC-MPC design to be in compliance with the dose limits in 10 CFR 72.106(b). The staff has reasonable assurance that the dose to any individual beyond the controlled area boundary from credible hypothetical accident conditions will not exceed these limits.

11.2.1 Accident Pressurization

11.2.1.1 Cause of Accident Pressurization

Accident pressurization assumes the failure of all of the fuel rods contained within the canister while at the maximum internal temperature. No credible events are identified that would result in either condition.

11.2.1.2 Consequences of Accident Pressurization

There are no storage conditions that are expected to lead to the rupture of all of the fuel rods and none that result in the assumed maximum temperature of 650°F. SAR Section 11.2.1 assumes, however, the hypothetical failure of all of the fuel rods in the TSC at a bounding temperature of 650°F to calculate the maximum internal pressure of 43.3 psig for the TSC. The SAR considers an internal pressure of 55 psig, which is bounding and conservative, for the TSC stress analysis. As evaluated in SER Section 3.3.5, the staff agrees with the SAR results that all stress margins are positive. This demonstrates that the TSC is structurally adequate to reasonably maintain confinement of radioactive material under the condition of accident pressurization. There are no radiological consequences for this accident.

11.2.2 Earthquake Event

11.2.2.1 Cause of Earthquake Event

It is possible that an earthquake could occur during the use of the NAC-MPC system.

11.2.2.2 Consequences of Earthquake Event

Earthquakes are natural phenomena that the NAC-MPC system might experience at an ISFSI. SAR Section 11.2.2 defines, at the top surface of the storage pad, the DBE motion of 0.25 g for the two horizontal components and 0.167 g for the vertical component. SER Section 3.4.3 reviews the SAR evaluation and concludes that, with adequate margins per ANSI/ANS-57.9, the cask will not slide or tip over under the DBE condition. On this basis, the staff concurs with the SAR conclusion that the VCC performance is not affected by the DBE and there are no radiological consequences for this natural phenomenon accident.

11.2.3 Explosion

11.2.3.1 Cause of Explosion

An explosion event is unlikely due to the administrative and security controls associated with an ISFSI operation. However, an explosion involving combustible materials at reactor sites is credible.

11.2.3.2 Consequences of Explosion

SAR Section 11.2.3 references the SAR Section11.2.6 analysis which demonstrates acceptable structural performance of the TSC subject to an external static pressure of 22 psig. On this basis, the staff concludes that, as a result of an explosion which exerts an equivalent static pressure of less than 22 psig on the canister, the NAC-MPC system will reasonably maintain confinement of radioactive material and that there are no radiological consequences for this explosion.

11.2.4 Failure of All Fuel Rods With a Subsequent Canister Breach

Prior to the issuance of NRC's ISG No. 3, an analysis of the dose consequence of a ground level canister breach, with 100% fuel rod failure, was required to demonstrate compliance with 10 CFR 72.106(a). However, this staff guidance requires that only credible accidents, and associated consequences, be evaluated against the requirements of 10 CFR Part 72. A credible accident is one which may lead to the following events: failure of the confinement boundary, transformation of the radioactive material into a dispersible form, release of such material from the cask, off-site dispersion of released materials, and/or an associated dose.

It is the staff's view that a ground level breach of the cask is a non-mechanistic failure since the confinement boundary is completely welded and tested to ANSI leaktight standards, and the stresses, temperatures, and pressures of the TSC are within the design basis limits under off-normal and hypothetical accident conditions. The TSC is vacuum dried and backfilled with helium gas prior to final canister closure, so there is no potential for an increase in the canister pressure or degradation of the cladding due to radiolytic decomposition or other adverse reactions. Further, the TSC is designed to be leaktight in accordance with ANSI N14.5-1997, as described in Section 7 of this SER.

The staff concludes that, under off-normal or hypothetical accident conditions, (1) an analysis of the dose consequence from this event is unnecessary since no discernable leakage of radioactive material from the TSC is credible (i.e., leaktight), (2) the dose consequence due to leakage of radioactive material from the all-welded canister is negligible, and (3) the requirements of 10 CFR 72.106(b) are met.

11.2.5 Fire Accident

11.2.5.1 Cause of Fire Accident

A major fire involving the NAC-MPC is unlikely, due to the absence of flammable materials in the vicinity of a spent fuel storage area. A transport vehicle fire during transfer of the VCC to the pad is considered credible.

11.2.5.2 Consequences of Fire Accident

A fire with an average flame temperature of 1475°F and duration of 8 minutes is postulated from the spillage and ignition of 50 gallons of combustible transporter fuel. The fire is assumed to spread along the ground and heat the air as it enters the cask. Solar insolation is applied during the fire since the fire is only assumed to occur at the base of the cask and a

heat load of 12.5 kW is applied to the cask walls. The initial temperature distribution of the transient is based on the normal storage conditions. Following the fire, the cask is cooled for 60 minutes using normal steady-state conditions at an ambient temperature of 75° F.

The applicant's evaluation showed that component temperatures remain less than the allowable temperatures, and thus, there are no significant radiological consequences for this accident. Local spalling of concrete could lead to a minor reduction in shielding effectiveness. A post-event inspection will determine the corrective actions necessary to ensure the cask remains within the design basis.

11.2.6 Flood

11.2.6.1 Cause of Flood

A flood event involving the NAC-MPC is considered credible. Possible natural events such as unusually high water from a river, dam break, seismic event, and severe weather are potential causes of floods.

11.2.6.2 Consequences of Flood

A flood event is a site-specific natural phenomenon. SAR Section 11.2.6 considers, however, the design basis flood conditions of a 50-foot depth of water having a velocity of 15 feet per second for the fully immersed NAC-MPC system. SER Section 3.4.1 reviews the SAR evaluation of the structural consequences of a flood to the NAC-MPC VCC and concurs with the SAR conclusion that the VCC will not overturn or slide, and the TSC will not suffer adverse consequences under the design basis flood conditions. On this basis, the staff agrees with the SAR analysis that there are no radiological consequences for this natural phenomenon accident.

11.2.7 Fresh Fuel Loading in the Canister

11.2.7.1 Cause of Fresh Fuel Loading in the Canister

Due to the administrative controls associated with candidate assembly selection, this event is not considered to be credible. However, it is evaluated to analyze the radiological consequences and bound potential mis-loadings.

11.2.7.2 Consequences of Fresh Fuel Loading in the Canister

The criticality evaluation assumes no burnup for the design basis assemblies and has determined adequate margin to criticality assuming the most reactive configuration and optimum moderation. Therefore, there are no radiological consequences associated with this event.

11.2.8 Full Blockage of Air Inlets and Outlets

11.2.8.1 Cause of Full Blockage of Air Inlets and Outlets

The likely cause of a full blockage of the VCC air inlets and outlets is a cask burial associated seismic event or landslide. The event is analyzed as a bounding condition and is not considered credible.

11.2.8.2 Consequences of Full Blockage of Air Inlets and Outlets

The applicant's evaluation assumed the sudden loss of convective cooling for the canister. The loss of convective cooling results in a sustained heat-up of the canister and concrete cask. Concrete temperatures exceed the thermal design criteria in less than 2 days. The maximum fuel temperature reached in the same approximately 2 day period is less than the short-term allowable temperature of 806°F.

Since the NAC-MPC retains its shielding performance, the radiological consequences of this event are low. Personnel dose associated with recovery actions to restore the air flow path is the most significant consequence and was estimated by the applicant to be about 50 mrem. Assuming debris is removed and the air flow path is restored in less than 2 days, the radiological consequences associated with this event are low.

11.2.9 Lightning

11.2.9.1 Cause of Lightning

Lightning is an expected natural phenomena to occur at an ISFSI site.

11.2.9.2 Consequences of Lightning

The applicant's analysis assumes that the lightning strikes the highest metal surface and proceeds to ground, resulting in heating along that path. The calculated increases in steel and concrete temperatures are small, and thus, there are no radiological consequences associated with this event.

11.2.10 Maximum Anticipated Heat Load (125°F Ambient Temperature)

11.2.10.1 Cause of Maximum Anticipated Heat Load

The assumed cause of this accident are weather events which subject the NAC-MPC to a 125°F ambient temperature with full solar insolance and maximum heat load.

11.2.10.2 Consequences of Maximum Anticipated Heat Load

An extreme environmental heat of 125°F is analyzed for the NAC-MPC with a maximum decay heat of 12.5 kW and a 12-hour insolation period of 2950 BTU/ft^2 and 1475 BTU/ft^2 for horizontal flat and curved surfaces, respectively. The evaluation shows that the component

temperatures are within allowable temperatures for the accident conditions and that the calculated concrete thermal stresses are also acceptable. Therefore, there are no radiological consequences associated with this event.

11.2.11 Storage Cask 6-Inch Drop

11.2.11.1 Cause of Storage Cask 6-Inch Drop

The loaded VCC is manipulated to the final storage destination at heights not to exceed 6-inches. Consequently, a 6-inch drop is considered credible. A failure involving the air pad or lifting jacks is postulated to be the cause of the accident.

11.2.11.2 Consequences of Storage Cask 6-Inch Drop

The VCC, which contains the loaded TSC, must be raised approximately 3 inches to install the inflatable air pads to allow it to be moved across the surfaces of the transporter and ISFSI pad. SAR Section 11.2.11 assumes a 6-inch drop of the VCC for calculating the bounding impact loads and evaluating consequences of the accident, including the crushing of the VCC concrete shell and permanent deformation of the TSC pedestal. The consequences of the loss of part of the air inlets are evaluated in SAR Section 11.1.1 and reviewed in SER Section 11.1.1. SER Section 3.3.7 reviews the SAR evaluation of the performance of the NAC-MPC system and notes that the accident might cause the VCC concrete shell to undergo an axial crush of 0.036 inch to result in a reduction of its shielding effectiveness. The SER also concurs with the SAR conclusion that the TSC is structurally adequate. On this basis, the staff concludes that the NAC-MPC system will reasonably maintain confinement of radioactive material under the 6-inch VCC drop accident, and thus, there are no radiological consequences for this event.

11.2.12 Tipover of the Vertical Concrete Cask

11.2.12.1 Cause of Tipover of the Vertical Concrete Cask

A tipover is possible in an earthquake that exceeds the DBE previously analyzed. There are no credible events expected to result in a cask tipover.

11.2.12.2 Consequences of Tipover of the VCC

Considering the structural performance, the staff concurs with the SAR assessment that no credible accidents, such as the 6-inch VCC drop and DBE, tornado, and flood natural phenomena will cause the VCC to tip over. To demonstrate the defense-in-depth design of the system, however, SER Section 3.3.8 reviews the SAR bounding case evaluation of the NAC-MPC under the tip over accident.

SER Sections 3.3.8.1 evaluates the SAR determination of deceleration g-loads applicable to the NAC-MPC components. SER Sections 3.3.8.2, 3.3.9, and 3.3.2 evaluate the SAR impact capabilities analysis for the TSC and support disk, fuel cladding, and RFA, respectively. The

evaluations support the SAR conclusion that the NAC-MPC does not suffer adverse structural consequences from the tipover accident and the VCC and TSC will maintain the design basis shielding, criticality control, and confinement performance requirements.

The SAR evaluates the radiological consequence in the hypothetical tipover accident and estimates the 1-meter and 5-meter dose rates, due to less shielding on the bottom of the cask, are approximately 156 and 1 rem/hr, respectively. The SAR notes that, following a tipover accident, supplemental shielding should be used for the exposed bottom of the tipped over VCC and TSC, and stringent access control must be applied to ensure that personnel do not enter the area of radiation shine from the exposed bottom of a tipped over cask. This radiological consequence is acceptable since the event is not considered to be credible.

11.2.13 Tornado and Tornado Driven Missiles

11.2.13.1 Cause of Tornado and Tornado Driven Missiles

It is possible that the NAC-MPC cask, which is placed on an unsheltered pad and subject to extreme weather, could be affected by the extreme winds associated with a tornado.

11.2.13.2 Consequences of Tornado and Tornado Driven Missiles

A tornado is a random weather event having a higher probability of occurrence at certain times of the year and in certain geographical areas. SAR Section 11.2.13 considers wind pressure and tornado driven missiles of the design basis tornado, for a maximum combined tornado wind speed of 360 mph, to evaluate the VCC for maintaining stability and providing protection of the TSC from missile penetration.

In SER Section 3.4.2, the staff reviews the SAR evaluation and concurs with its conclusion that the tornado wind pressure and tornado driven missiles are not capable of overturning the cask or penetrating the boundary established by the concrete cask to affect the performance of the TSC. The staff also concurs with the SAR evaluation that a penetration of about 6 inches into the concrete shield is possible and the resulting local surface radiation dose rate would not cause a noticeable dose rate increase at the site boundary. On this basis, the staff concludes that the system will reasonably maintain confinement of radioactive material when subject to tornado wind and tornado driven missiles and that the radiological consequences are small.

11.3 Criticality

As discussed in SER Section 6, the applicant has shown, and the staff has verified, that the spent fuel remains subcritical ($k_{eff} < 0.95$) under all credible conditions from normal, off-normal, and postulated accident events. The design basis off-normal and accident events do not adversely affect the design features important to criticality safety. Therefore, based on the information provided in the SAR, the staff concludes that the NAC-MPC system design meets the "double contingency" requirements of 10 CFR 72.124(a).

11.4 Post-Accident Recovery

Section 11.2 of the SAR discusses corrective actions for each accident identified in Section 11.2. The SAR did not identify a design basis accident that would affect the canister confinement boundary or significantly damage the cask system at a level that could result in undue risk to public health and safety.

The staff reviewed the design basis accident analyses with respect to post-accident recovery and found them to be acceptable. The staff has reasonable assurance that the site licensee can recover the NAC-MPC from the analyzed design basis accidents and that the generic corrective actions outlined in the SAR are appropriate to protect public health and safety.

11.5 Instrumentation

Because of the passive nature of the NAC-MPC system, no instrumentation and control systems are needed to monitor SSCs important to safety. Therefore, there are no instrumentation and control systems that must remain operational under accident conditions. The confinement boundary contains no external penetrations for pressure monitoring or overpressure protection. Since the TSC uses an entirely welded redundant closure system and, under normal and off-normal conditions, there are no anticipated mechanisms that would cause weld failure, no direct monitoring of the closure is required.

11.6 Evaluation Findings

F11.1 The SSCs of the NAC-MPC system are adequate to prevent accidents and to mitigate the consequences of accidents and natural phenomena events that do occur.

F11.2 The spacing of casks, discussed in Section 1.4 of the SAR and included as a design specification (see Section 4.5 of the TS), will ensure accessibility of the equipment and services required for emergency response.

F11.3 Table 12-1 of this SER lists the TS for the NAC-MPC. These TS are further discussed in Section 12 of the SER.

F11.4 The applicant has evaluated the NAC-MPC to demonstrate that it will reasonably maintain confinement of radioactive material under credible accident conditions.

F11.5 An accident or natural phenomena event will not preclude the safe recovery of the NAC-MPC cask.

F11.6 The spent fuel will be maintained in a subcritical condition under accident conditions.

F11.7 The applicant has evaluated off-normal and design basis accident conditions to demonstrate with reasonable assurance that the NAC-MPC radiation shielding and confinement features are sufficient to meet the requirements in 10 CFR 72.104(a) and 10 CFR 72.106(b).

F11.8 No instrumentation or control systems are required to remain operational under accident conditions.

F11.9 The staff concludes that the accident design criteria for the NAC-MPC are in compliance with 10 CFR Part 72 and the accident design and acceptance criteria have been satisfied. The applicant's accident evaluation of the cask adequately demonstrates that it will provide for safe storage of spent fuel during credible accident situations. This finding is reached on the basis of a review that considered independent confirmatory calculations, the regulation itself, appropriate regulatory guides, applicable codes and standards, and accepted engineering practices.

12.0 CONDITIONS FOR CASK USE —TECHNICAL SPECIFICATIONS

The purpose of the review of the conditions for cask use is to determine whether the applicant has fully evaluated the TS and to ensure that the SER incorporates any additional operating controls and limits that the staff deems necessary.

12.1 Conditions for Use

The conditions for use of the NAC-MPC system are fully defined in the CoC and the TS which are appended to it.

12.2 Technical Specifications

Table 12-1 lists the TS for the NAC-MPC system. The staff has appended these TS to the CoC for the NAC-MPC.

TABLE 12-1
NAC-MPC SYSTEM TECHNICAL SPECIFICATIONS

NUMBER	TECHNICAL SPECIFICATION
1.0	USE AND APPLICATION
1.1	Definitions
1.2	Logical Connectors
1.3	Completion Times
1.4	Frequency
2.0	FUNCTIONAL AND OPERATING LIMITS
2.1	Functional and Operating Limits
2.2	Functional and Operational Limit Violations
Table 2-1	Fuel Assembly Limits
Table 2-2	INTACT FUEL ASSEMBLY Characteristics
3.0	LIMITING CONDITION FOR OPERATION (LCO) APPLICABILITY and SURVEILLANCE REQUIREMENT (SR) APPLICABILITY
3.1	NAC-MPC SYSTEM Integrity
3.1.1	Reserved
3.1.2	CANISTER Vacuum Drying Pressure
3.1.3	CANISTER Helium Backfill Pressure
3.1.4	CANISTER Helium Leak Rate
3.1.5	CANISTER Maximum Time in Vacuum Drying
3.1.6	CANISTER Maximum Time in TRANSFER CASK
3.1.7	Fuel Cooldown Requirements
3.1.8	CONCRETE CASK Maximum Lifting Height
3.1.9	TRANSFER CASK Minimum Operating Temp.
3.1.10	CANISTER Removal from the CONCRETE CASK
Table 3-1	CANISTER Limits
3.2	NAC-MPC SYSTEM Radiation Protection
3.2.1	NAC-MPC SYSTEM Average Surface Dose Rates
Figure 3-1	CONCRETE CASK Surface Dose Rate Measurement
3.2.2	CANISTER Surface Contamination

TABLE 12-1 (continued)
NAC-MPC SYSTEM TECHNICAL SPECIFICATIONS

NUMBER	TECHNICAL SPECIFICATION
4.0	DESIGN FEATURES
4.1	Site
4.2	Storage Features
4.3	Codes and Standards
4.4	Site Specific Parameters and Analyses
4.5	Design Specifications
Table 4-1	List of ASME Code Exceptions for CANISTER
5.0	ADMINISTRATIVE CONTROLS
5.1	NAC-MPC SYSTEM Training
5.2	Dry Run Training
5.3	Special Requirements for First NAC-MPC SYSTEM Placed in Service
5.4	Programs
5.4.1	CONCRETE CASK Thermal Monitoring Program

12.3 Evaluation Findings

F12.1 Table 12-1 of the SER lists the TS for the NAC-MPC. These TS are further discussed in Section 12 of the SAR and are part of the CoC.

F12.2 The staff concludes that the conditions for use of the NAC-MPC identify necessary TS to satisfy 10 CFR Part 72 and that the applicable acceptance criteria have been satisfied. The TS provide reasonable assurance that the cask will allow safe storage of spent fuel. This finding is reached on the basis of a review that considered the regulation itself, appropriate regulatory guides, applicable codes and standards, and accepted practices.

13.0 QUALITY ASSURANCE

The purpose of this review and evaluation is to determine whether NAC has a QA program that complies with the requirements of 10 CFR Part 72, Subpart G.

13.1 Areas Reviewed

QA Organization
QA Program
Design Control
Procurement Document Control
Instructions, Procedures, and Drawings
Document Control
Control of Purchased Material, Equipment, and Services
Identification and Control of Materials, Parts, and Components
Control of Special Processes
Licensee Inspection
Test Control
Control of Measuring and Test Equipment
Handling, Storage, and Shipping Control
Inspection, Test, and Operating Status
Nonconforming Materials, Parts, or Components
Corrective Action
QA Records
Audits

NUREG-1536 provides the criteria for evaluating the above 18 areas. As indicated in Section 13.1 of the SAR, the NRC has issued a QA program approval for activities conducted under Subpart H of 10 CFR Part 71. Based on the review of the QA program described in the SAR and previous NRC determinations regarding NAC's 10 CFR Part 72 QA program, the staff has determined that it meets the requirements of Subpart G of 10 CFR Part 72. The NRC has also performed inspections of NAC's implementation of the 10 CFR Part 71 and 72 QA programs, most recently in June 1998. While this evaluation determined that the QA program is acceptable, continued proper implementation of the QA program will continue to be assessed during future NRC inspections.

13.2 Evaluation Findings

F13.1 The QA program describes the requirements, procedures, and controls that, when properly implemented, comply with the requirements of 10 CFR Part 72, Subpart G and 10 CFR Part 21, "Reporting of Defects and Noncompliance."

F13.2 The structure of the organization and assignment of responsibility for each activity ensure that designated parties will perform the work to achieve and maintain specified quality requirements.

F13.3 Conformance to established requirements will be verified by qualified personnel and groups not directly responsible for the activity being performed. These personnel and groups report through a management hierarchy which grants the necessary authority and organizational freedom and provides sufficient independence from economic and scheduling influences.

F13.4 The QA program is well-documented and provides adequate control over activities affecting quality, as well as SSCs important to safety, consistent with their relative importance to safety (graded approach).

F13.5 NAC's QA program complies with the applicable NRC regulations and can be implemented for the design, fabrication, testing, modification, and use of the NAC-MPC system.

F13.6 This SAR can be referenced without further QA review in a license application to receive and store spent fuel under 10 CFR Part 72, provided that the applicant applies its NRC-approved QA program meeting the requirements of 10 CFR Part 50, Appendix B, to the design, construction, and use of SSCs that are important to safety for a spent fuel storage installation.

14.0 DECOMMISSIONING

The purpose of the review of the conceptual decommissioning plan for the NAC-MPC system is to ensure that it provides reasonable assurance that the owner of the cask can conduct decontamination and decommissioning in a manner that adequately protects the health and safety of the public. Nothing in this review considers, or involves the review of, ultimate disposal of spent nuclear fuel.

14.1 Decommissioning Considerations

The conceptual decommissioning plan for the NAC-MPC is provided in Section 2.4 of the SAR. While NAC clearly anticipates that the TSC could be used as part of a final geologic disposal system, the ability to decommission the TSC is also considered. Table 2.4.1 of the SAR provides the activity concentrations of the major radiation sources in the VCC and TSC which NAC has determined would exist after 40 years of activation by 36 design basis fuel assemblies stored in the NAC-MPC system. The material activation results presented in SAR Table 2.4.1 confirm that total system activation is low for all components. Therefore, the canister and concrete cask could be disposed in a near-surface facility as low specific activity material.

NAC determined that the VCC and TSC can be decommissioned using standard industry practices. Activated steel components can be decontaminated using existing mechanical or chemical methods.

14.2 Evaluation Findings

F14.1 The NAC-MPC system design includes adequate provisions for decontamination and decommissioning. As discussed in Section 2.4 of the SAR, these provisions include facilitating decontamination of the NAC-MPC, if needed; storing the remaining components, if no waste facility is expected to be available; and disposing of any remaining low-level radioactive waste.

F14.2 Section 2.4 of the SAR also presents information concerning the proposed practices and procedures for decontaminating the cask and disposing of residual radioactive materials after all spent fuel has been removed. This information provides reasonable assurance that the applicant will conduct decontamination and decommissioning in a manner that adequately protects public health and safety.

F14.3 The staff concludes that the decommissioning considerations for the NAC-MPC are in compliance with 10 CFR Part 72. This evaluation provides reasonable assurance that the NAC-MPC will allow safe storage of spent fuel. This finding is reached on the basis of a review that considered the regulation itself, appropriate regulatory guides, applicable codes and standards, and accepted engineering practices.

CONCLUSIONS

Based on the statements and representations contained in the application, as supplemented, and the conditions listed above, we have concluded that the Model No. NAC-MPC spent fuel storage cask system meets the requirements of Part 72.

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