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May 23, 2000

U.S. Nuclear Regulatory Commission
11555 Rockville Pike
Rockville, MD 20852-2738

Attn: Document Control Desk

Subject: Docket No. 72-1025

Submittal of NAC-MPC FSAR, Revision 0 (TAC No. L22907)

Reference:

1. Submittal of Changed Pages for Revision 5 of the NAC-MPC SAR, NAC International, January 5, 2000
2. Certificate of Compliance for the NAC International, Inc, Multi-Purpose Canister (MPC) System, United States Nuclear Regulatory Commission, March 17, 2000

In accordance with the requirements of 10 CFR 72.248(a)(1), NAC herewith provides the NAC-MPC Final Safety Analysis Report (FSAR), Revision 0, for the NAC-MPC System approved by Reference 2. This FSAR is based on the safety analysis report submitted with the initial application and includes all changes through Reference 1, which were incorporated during the review process. The FSAR reflects all of the requirements contained in Reference 2.

If you have any comments or questions, please contact me or Jim Ballowe at (770) 447-1144.

Sincerely,

T.C. Thompson

Thomas C. Thompson
Director, Licensing & Competitive Assessment
Engineering & Design Services

cc: Joseph McCumber DE&S

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NAC-MPC

FINAL SAFETY ANALYSIS REPORT

for the

NAC Multi-Purpose Canister System

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APRIL 2000

REVISION 0

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1.0 GENERAL DESCRIPTION

NAC International Inc. (NAC) has designed a Multi-Purpose Canister system (NAC-MPC) for the long-term storage of spent nuclear fuel. The NAC-MPC system consists of a transportable storage canister, vertical concrete cask, and a transfer cask.

The transportable storage canister is designed and fabricated to meet the requirements for transport in the NAC Storable Transport Cask (NAC-STC) and to be compatible with the U.S. Department of Energy MPC Design Procurement Specification so as not to preclude the possibility of permanent disposal in a deep Mined Geological Disposal System.

In long-term storage, the transportable storage canister is installed in a vertical concrete cask, which provides passive radiation shielding and natural convection cooling. The vertical concrete storage cask also provides protection during storage for the transportable storage canister under adverse environmental conditions. The NAC-MPC employs a double-welded closure design to preclude loss of contents and to preserve the general health and safety of the public during long-term storage of spent fuel.

The transfer cask is used to move the transportable storage canister from the work stations where the canister is loaded and closed to the vertical concrete cask. It is also used to transfer the canister from the vertical concrete cask to the NAC-STC for transport.

This Safety Analysis Report demonstrates the ability of the NAC-MPC System to satisfy the Nuclear Regulatory Commission (NRC) requirements for the storage of spent fuel, as prescribed by 10 CFR 72.

This chapter provides a general description of the major components of the system and a description of the system operation. The terminology used throughout this report is summarized in Table 1-1. Table 1-2 provides a compliance matrix to the regulatory requirements and acceptance criteria specified in NUREG-1536. This matrix describes how the NAC-MPC Safety Analysis Report (SAR) complies with each requirement and criterion listed in NUREG-1536. Table 12A4-1 provides a list of the exceptions from the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code) for the canister.

Table 1-1 Terminology

NAC-STC Cask	The licensed spent-fuel transport cask consisting of a spent fuel storable transport cask body with dual closure lids and energy-absorbing impact limiters (Certificate of Compliance No. 71-9235).
Confinement System	The components of the transportable storage canister intended to retain the radioactive material during storage.
Transportable Storage Canister (Canister)	The stainless steel cylindrical shell, bottom end plate, shield lid, and structural lid that holds the spent fuel in the canister basket.
Contents	Up to 36 intact Yankee Class pressurized-water reactor (PWR) spent fuel assemblies and Reconfigured Fuel Assemblies to a maximum total contents weight of 30,600 pounds in the transportable storage canister.
Yankee Class Spent Fuel	Fuel that includes United Nuclear Type A and Type B, Combustion Engineering Type A and Type B, Exxon-ANF Type A and Type B, and Westinghouse Type A and Type B spent fuel assemblies.
Reconfigured Fuel Assembly (RFA)	A stainless steel container having the same external dimensions as a standard Yankee Class spent fuel assembly that ensures criticality control geometry and which permits gaseous and liquid media to escape while minimizing dispersal of gross particulates. The reconfigured fuel assembly may contain a maximum of 64 intact fuel rod, damaged fuel rods or fuel debris from any type of Yankee Class spent fuel assembly.
Intact Fuel Assembly	A fuel assembly without known or suspected cladding defects greater than a pinhole leak or a hairline crack and which can be handled by normal means. A partial fuel assembly is a fuel assembly from which fuel rods are missing. A partial fuel assembly shall not be classified as an intact fuel assembly unless solid Zircaloy or stainless steel rods are used to displace an amount of water equal to that displaced by the original fuel rod(s).

Table 1-1 Terminology (Continued)

Intact Fuel Rod	A fuel rod without known or suspected cladding defects greater than a pinhole leak or a hairline crack.
Damaged Fuel Assembly	A fuel assembly with known or suspected cladding defects greater than a hairline crack or a pinhole leak.
Damaged Fuel Rod	A fuel rod with known or suspected cladding defects greater than a hairline crack or a pinhole leak.
Fuel Debris	Fuel in the form of particles, loose pellets and fragmented rods or assemblies.
Canister Basket	The structure placed in the transportable storage canister to support the fuel assemblies (fuel basket).
- **Support Disk**	A circular stainless steel plate with square holes machined in a symmetrical pattern that provides the primary lateral load-bearing component of the canister basket.
- **Heat Transfer Disk**	A circular aluminum plate with square holes machined in a symmetrical pattern. The heat transfer disk enhances heat transfer in the fuel basket.
- **Fuel Tube**	A stainless steel tube having a square cross-section and BORAL neutron poison material on its exterior surfaces.
- **Tie Rod**	A stainless steel rod used to align the supports disks and heat transfer disks in the fuel basket structure.
- **Split Spacer**	Spacers installed on the tie rod between the support disks to properly position, and provide axial support for, the support disks and the heat transfer disks.

Table 1-1 Terminology (Continued)

Shield Lid	The primary confinement boundary for the canister. It is located directly above the canister basket.
- Drain Port	A penetration located in the shield lid to permit draining of the canister cavity.
- Vent Port	A penetration located in the shield lid to aid in draining and backfilling the canister cavity.
- Port Cover	The stainless steel covers that close the vent and drain ports, which are welded in place following draining, drying, and backfilling operations.
- Quick Disconnect	The quick-disconnect valved nipple used in the vent and drain ports to facilitate operations.
Structural Lid	The secondary confinement boundary for the canister. The structural lid provides the lifting point for the loaded canister.
Vertical Concrete Cask (Concrete Cask) (Storage Cask)	A reinforced concrete cylinder closed at the top end by a shield plug and lid that holds the transportable storage canister during storage. The vertical concrete cask is formed around a steel inner liner and base.
- Shield Plug	A thick carbon steel plug installed in the top end of the storage cask to reduce skyshine radiation. The shield plug contains a one-inch thick neutron shield.
- Lid	A thick carbon steel bolted closure for the storage cask. The lid precludes access to the canister and provides additional radiation shielding.

Table 1-1 Terminology (Continued)

- Liner	A thick carbon steel shell that forms the annulus of the concrete storage cask. The liner serves as the inner form during concrete pouring and provides radiation shielding of the canister contents.
- Base	A carbon steel weldment that contains the inlet air vents, the storage cask jacking points, and the pedestal that supports the canister inside the storage cask.
Transfer Cask	A shielded lifting device for the empty and loaded canister. It is used for the vertical transfer of the canister between work stations and the storage cask or the transport cask. The transfer cask incorporates bottom doors that permit the vertical loading of the storage and transport casks.
- Lifting Trunnions	Two carbon steel trunnions used to lift and move the transfer cask.
Adapter Plate	A carbon steel plate that attaches to the top of the transport or storage cask to facilitate the installation and alignment of the transfer cask. It also provides the operating mechanism for the transfer cask bottom doors.
Margin of Safety	An analytically determined value defined as the "factor of safety" minus 1. Factor of safety is also analytically determined and is defined as the allowable stress of a material divided by its actual (calculated) stress.

Table 1-2 NUREG-1536 Compliance Matrix

Chapter 1 – General Description			
Area	**Requirement**	**Acceptance Criteria**	**Description of Compliance**
1. General Description and Operational Features	The application must present a general description and discussion of the DCSS, with special attention to design and operating characteristics, unusual or novel design features, and principal safety considerations. [10 CFR Part 72.24(b)]	The applicant should provide a broad overview and a general, non-proprietary description (including illustrations) of the DCSS, clearly identifying the functions of all components and providing a list of those components classified by the applicant as being "important to safety."	A general description of the system is provided in Section 1.2.
2. Drawings	Structures, systems, and components (SSCs) important to safety must be described in sufficient detail to enable reviewers to evaluate their effectiveness. [10 CFR Part 72.24(c)(3)]	The applicant should provide non-proprietary drawings of the storage system, of sufficient detail, that an interested party can ascertain its major design features and general operations.	Drawings of the system are provided in Section 1.5. Safety classifications are provided in Table 2.3-1
3. DCSS Contents	The applicant must provide specifications for the contents expected to be stored in the DCSS (normally spent fuel). These specifications may include, but not be limited to, type of spent fuel (i.e., boiling-water reactor [BWR], pressurized-water reactor [PWR], or both), maximum allowable enrichment of the fuel before any irradiation, burnup (i.e., megawatt-days/metric ton Uranium), minimum acceptable cooling time of the spent fuel before storage in the DCSS (aged at least 1 year), maximum heat designed to be dissipated, maximum spent fuel loading limit, condition of the spent fuel (i.e., intact assembly or consolidated fuel rods), weight and nature of non-spent fuel contents, and inert atmosphere requirements. [10 CFR Part 72.2(a)(1) and 10 CFR Part 72.236(a)]	The applicant should characterize the fuel and other radioactive wastes expected to be stored in the DCSS. If the potential exists that the DCSS will be used to store degraded fuel, the SAR should include a discussion of how the sub-criticality and retrievability requirements will be maintained.	A description of the contents to be stored is presented in Section 1.2.3.

Table 1-2 NUREG-1536 Compliance Matrix (Continued)

Chapter 1 – General Description			
Area	**Requirement**	**Acceptance Criteria**	**Description of Compliance**
4. Qualifications of the Applicant	The application must include the technical qualifications of the applicant to engage in the proposed activities. Qualifications should include training and experience. [10 CFR Part 72.24(j), 10 CFR Part 72.28(a)]	The reviewer should ensure that the applicant has clearly identified the roles and responsibilities that the DCSS designer, vendor, and other agents, such as potential licensees, fabricators, and contractors will have in the review process. Verify that the applicant has provided clear evidence demonstrating that they are qualified to engage in the proposed activities. In addition, verify that the applicant has delineated the responsibilities for all those who will be involved in the construction and operation of the DCSS if known. The reviewer should ensure that the applicant has specifically defined activities, which they will not perform.	Applicant qualifications are discussed in Section 1.3.
5. Quality Assurance	The safety analysis report (SAR) must include a description of the applicant's quality assurance (QA) program, with reference to implementing procedures. This description must satisfy the requirements of 10 CFR Part 72, Subpart G, and must be applied to DCSS SSC that are important to safety throughout all design, fabrication, construction, testing, operations, modifications and decommissioning activities. These implementing procedures need not be explicitly included in the application. [10 CFR Part 72.24(n)]	Verify that the applicant has described the proposed QA program, citing the applicable implementing procedures. This description should satisfy all requirements of 10 CFR Part 72, Subpart G, that apply to the design, fabrication, construction, testing, operation, modification, and decommissioning of the DCSS SSCs that are important to safety.	Applicant QA program is presented in Chapter 13.
6. Consideration of 10 CFR Part 71 Requirements Regarding Transportation	If the DCSS under consideration has previously been reviewed and certified for use as a transportation cask, the application must include a copy of the Certificate of Compliance issued for the DCSS under 10 CFR Part 71, including drawings and other documents referenced in the certificate. [10 CFR 72.230(b)]	If the DCSS under review has previously been evaluated for use as a transportation cask, the submittal should include the Part 71 Certificate of Compliance and associated documents.	The transport SAR and Certificate of Compliance are discussed in Section 1.2.1.

Table 1-2 NUREG-1536 Compliance Matrix (Continued)

Chapter 2 – Principal Design Criteria			
Area	**Requirement**	**Acceptance Criteria**	**Description of Compliance**
1. Structures, Systems, and Components Important to Safety	The applicant must identify all SSC that are important to safety, and describe the relationships of non-important to safety SSC on overall DCSS performance. [10 CFR 72.24(c)(3) and 72.44(d)] The applicant must specify the design bases and criteria all SSC that are important to safety. [10 CFR 72.24(c)(1), 72.24(c)(2), 72.120(a), and 72.236(b)]	The applicant should discuss the general configuration of the DCSS, and should provide an overview of specific components and their intended functions. In addition, the applicant should identify those components deemed to be important to safety, and should address the safety functions of those components in terms of how they meet the general design criteria and regulatory requirements discussed above. Additional information concerning specific functional requirements for individual DCSS components are addressed in the subsequent chapters of this SRP.	The safety classification of system components are described in Table 2.3-1. The design bases and criteria for the system are specified in Table 2-1. Detailed design criteria are presented in Section 2.2

Table 1-2 NUREG-1536 Compliance Matrix (Continued)

Chapter 2 - Principal Design Criteria			
Area	**Requirement**	**Acceptance Criteria**	**Description of Compliance**
2. Design Bases for Structures, Systems, and Components Important to Safety **a. Spent Fuel Specifications**	The applicant must provide the range of specifications for the spent fuel to be stored in the DCSS. These specifications should include, but are not to be limited to: the type of spent fuel (i.e., boiling-water reactor [BWR], pressurized-water reactor [PWR], or both); content, weight, dimensions and configurations of the fuel; maximum allowable enrichment of the fuel before any irradiation; maximum fuel burnup (i.e., megawatt-days/MTU); minimum acceptable cooling time of the spent fuel before storage in the DCSS (aged at least 1 year); maximum heat load to be dissipated; maximum spent fuel elements to be loaded; spent fuel condition (i.e., intact assembly or consolidated fuel rods); and any inerting atmosphere requirements. [10 CFR 72.2(a)(1) and 72.236(a)]	Detailed descriptions of each of the items listed below are generally found in specific sections of the SAR; however, a brief description of these areas, including a summary of the analytical techniques used in the design process, should also be captured in Section 2 of the SAR. This description gives reviewers a perspective on how specific DCSS components interact to meet the regulatory requirements of 10 CFR Part 72. This discussion should be non-proprietary, since it may be used to familiarize interested persons with the design features and bounding conditions of operation of a given DCSS. The applicant should define the range and types of spent fuel or other radioactive materials that the DCSS is designed to store. In addition, these specifications should include, but are not to be limited to, the type of spent fuel (i.e., boiling-water reactor [(BWR], pressurized-water reactor [PWR], or both), weights of the stored materials, dimensions and configurations of the fuel, maximum allowable enrichment of the fuel before any irradiation, burnup (i.e., megawatt-days/MTU), minimum acceptable cooling time of the spent fuel before storage in the DCSS (aged at least 1 year), maximum heat designed to be dissipated, maximum number of spent fuel elements, condition of the spent fuel (i.e., intact assembly or consolidated fuel rods), inerting atmosphere requirements, and the maximum amount of fuel permitted for storage in the DCSS. For DCSSs that will be used to store radioactive materials other than spent fuel, that is, activated components associated with a spent fuel assembly (e.g., control rods, BWR fuel channels), the applicant should specify the types and amounts of radionuclides, heat generation and the relevant source strengths and radiation energy spectra permitted for storage in the DCSS.	Specifications of the spent fuel contents are provided in Section 2.1. Specific physical parameters of the fuel are listed in Table 2.1-1.

Table 1-2 NUREG-1536 Compliance Matrix (Continued)

Chapter 2 - Principal Design Criteria			
Area	**Requirement**	**Acceptance Criteria**	**Description of Compliance**
2. Design Bases for Structures, Systems, and Components Important to Safety **b. External Conditions**	The design bases for SSC important to safety must reflect an appropriate consideration of environmental conditions associated with normal operations, as well as design considerations for both normal and accident conditions and the effects of natural phenomena events. [10 CFR 72.122(b)]	The SAR should define the bounding conditions under which the DCSS is expected to operate. Such conditions include both normal and off-normal environmental conditions, as well as accident conditions. In addition, the applicant should consider the effects of natural events, such as tornadoes, earthquakes, floods, and lightning strikes. The effects of such events are addressed in individual chapters of the SRP (e.g., the effects of an earthquake on the DCSS structural components are addressed in Chapter 3, "Structural Analysis").	The environmental conditions and natural phenomena considered as design bases are described in Section 2.2.
3. Design Criteria for Safety Protection Systems **a. General**	The DCSS must be designed to safely store the spent fuel for a minimum of 20 years and to permit maintenance as required. [10 CFR 72.236(g)] SSC important to safety must be designed, fabricated, erected, and tested to quality standards commensurate with the importance to safety of the function to be performed. [10 CFR 72.122(a)] The applicant must identify all codes and standards applicable to the SSC. [10 CFR 72.24(c)(4)]	The SAR should define an expected lifetime for the cask design. The staff has accepted a minimum of 20 years as consistent with the licensing period. The applicant should also briefly describe the proposed quality assurance (QA) program, and applicable industry codes and standards that will be applied to the design, fabrication, construction, and operation of the DCSS. In establishing normal and off-normal conditions applicable to the design criteria for DCSS designs, applicants should account for actual facility operating conditions. Design considerations should therefore reflect normal operational ranges, including any seasonal variations or effects.	The codes and standards of design and construction of the system are specified in Section 3.1.2.

Table 1-2 NUREG-1536 Compliance Matrix (Continued)

Chapter 2 - Principal Design Criteria			
Area	**Requirement**	**Acceptance Criteria**	**Description of Compliance**
3. Design Criteria for Safety Protection Systems **b. Structural**	SSC that are important to safety must be designed to accommodate the combined loads of normal operations, accidents, and natural phenomena events with an adequate margin of safety. [10 CFR 72.24(c)(3), 72.122(b), and 72.122(c)] The design-basis earthquake must be equivalent to or exceed the safe shutdown earthquake of a nuclear plant at sites evaluated under 10 CFR Part 100. [10 CFR 72.102(f)] The DCSS must maintain confinement of radioactive material within the limits of 10 CFR Part 72 and Part 20, under normal, off-normal, and credible accident conditions. [10 CFR 72.236(l)] The DCSS must be designed and fabricated so that the spent fuel is maintained in a subcritical condition all under all credible normal, off-normal, and accident conditions. [10 CFR 72.124(a) and 72.236(c)] The spent fuel cladding must be protected during storage against degradation that leads to gross ruptures, or the fuel must be otherwise confined such that degradation of the fuel during storage will not pose operational safety problems with respect to its removal from storage. [10 CFR 72.122(h)(1)] Storage systems must be designed to allow ready retrieval of spent fuel waste for further processing or disposal. [10 CFR 72.122(l)]	The SAR should define how the DCSS structural components are designed to accommodate combined normal, off-normal, and accident loads, while protecting the DCSS contents from significant structural degradation, criticality, and loss of confinement, while preserving retrievability. This discussion is generally a summary of the analytical techniques and calculational results from the detailed analysis discussed in SAR Section 3 and should be presented in a non-proprietary forum.	A discussion of the structural design criteria are presented in Section 2.2. Combined loadings are addressed specifically in Section 2.2.5 and in Tables 2.2-2 and 2.2-3. The design-basis earthquake is specified in Section 2.2.3 in accordance with 10 CFR 72.102 criteria. Because the system maintains adequate margins of safety during normal (Section 3.4.4.1), off-normal (Section 11.1) and accident condition (Section 11.2) events, confinement of the radioactive material is assured. Because the system maintains adequate structural margins of safety during normal, off-normal and accident condition events, criticality control is assured based on the analyses presented in Chapter 6. The maximum allowable cladding temperatures are specified in Tables 2-1 and 4.1-3. The temperature results for the fuel cladding listed in Table 4.1-4 show that the allowable cladding temperatures are not exceeded. Therefore, the fuel cladding is protected against degradation during storage. As described in Section 1.2, the system is designed to be readily transported as necessary for further processing or disposal.

Table 1-2 NUREG-1536 Compliance Matrix (Continued)

Chapter 2 - Principal Design Criteria			
Area	**Requirement**	**Acceptance Criteria**	**Description of Compliance**
3. Design Criteria for Safety Protection Systems **c. Thermal**	Each spent fuel storage or handling system must be designed with a heat removal capability having testability and reliability consistent with its importance to safety. [10 CFR 72.128(a)(4)] The DCSS must be designed to provide adequate heat removal capacity without active cooling systems. [10 CFR 72.236(f)]	The applicant should provide a general discussion of the proposed heat removal mechanisms, including the reliability and verifiability of such mechanisms and any associated limitations. All heat removal mechanisms should be passive and independent of intervening actions under normal and off-normal conditions.	The testability of the heat removal capability of the storage system is described in Section 2.3.3.2. The reliability of the heat removal system is demonstrated in Chapter 4, and operating limits are established in Chapter 12 consistent with the temperature monitoring and routine surveillance described in Section 2.3.3.2 to ensure continued safe operation. As shown in Table 4.1-4, the storage system provides adequate heat removal through the passive cooling design features described in Section 1.2.1.2.

Table 1-2 NUREG-1536 Compliance Matrix (Continued)

Chapter 2 - Principal Design Criteria			
Area	**Requirement**	**Acceptance Criteria**	**Description of Compliance**
3. Design Criteria for Safety Protection Systems **d. Shielding/Confinement/ Radiation Protection**	The proposed DCSS design must provide radiation shielding and confinement features that are sufficient to meet the requirements of 10 CFR 72.104 and 72.106. [10 CFR 72.126(a), 72.128(a)(2), 72.128(a)(3), and 72.236(d)] During normal operations and other 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 as a result of exposure to (1) planned discharges to the general environment of radioactive materials except radon and its decay products, (2) direct radiation from operations of the ISFSI or monitored retrievable storage (MRS), and (3) any other radiation from uranium fuel cycle operations within the region. [10 CFR 72.24(d), 72.104(a), and 72.236(d)] Any individual located at or beyond the nearest boundary of the controlled area shall not receive a dose greater than 5 rem to the whole body or any organ from any design-basis accident. The minimum distance from the spent fuel handling and storage facilities to the nearest boundary of the controlled area shall be 100 meters. [10 CFR 72.24(d), 72.24(m), 72.106(b), and 36(d)] The DCSS must be designed to provide redundant sealing of confinement systems. [10 CFR 72.236(e)] Storage confinement systems must have the capability for continuous monitoring in a manner such that the licensee will be able to determine when corrective action needs to be taken to maintain safe storage conditions. [10 CFR 72.122(h)(4) and 72.128(a)(1)] The DCSS design must include inspections, instrumentation and/or control (I&C) systems to monitor the SSC that are important to safety over anticipated ranges for normal and off-normal operation. In addition, the applicant must identify those control systems that must remain operational under accident conditions. [10 CFR 72.122(i)]	The applicant should describe those features of the cask that protect occupational workers and members of the public against direct radiation dosages and releases of radioactive material, and minimize the dose after any off-normal or accident conditions.	The confinement design features are described in Section 2.3.2.1, while the radiation shielding design features are described in Section 2.3.5. Section 10.4 presents the necessary minimum site boundary distances from an array of loaded storage systems to meet the controlled area dose limits. As stated in Section 10.2.2, there is no postulated accident condition that would result in a release of radioactive materials. Therefore, the accident dose limit is met. The redundant sealing features of the confinement system are presented in Section 2.3.2.1. As described in Section 2.3.1, the system is passive and can operate through all postulated normal, off-normal, and accident events while maintaining safe storage conditions for the fuel. As specified in Appendix 12A, Section 3.1.7, temperature monitoring is utilized to verify that the systems operating conditions are maintained. Appendix 12A, Section 3.1.7 and Section 12.4 specify the surveillance requirements for the system under normal conditions and after an accident, respectively. These activities are specified to ensure that the system is operated within its design parameters at all times.

Table 1-2 NUREG-1536 Compliance Matrix (Continued)

Chapter 2 - Principal Design Criteria			
Area	**Requirement**	**Acceptance Criteria**	**Description of Compliance**
3. Design Criteria for Safety Protection Systems **e. Criticality**	Spent fuel transfer and storage systems must be designed to remain subcritical under all credible conditions. [10 CFR 72.124(a) and 72.236(c)] When practicable, the DCSS must be designed on the basis of favorable geometry, permanently fixed neutron-absorbing materials (poisons), or both. Where solid neutron-absorbing materials are used, the design shall allow for positive means to verify their continued efficacy. [10 CFR 72.124(b)]	The SAR should address the mechanisms and design features that enable the DCSS to maintain spent fuel in a subcritical condition under normal, off-normal, and accident conditions.	The criticality safety design criteria for the system are presented in Section 2.3.4.

Table 1-2 NUREG-1536 Compliance Matrix (Continued)

Chapter 2 - Principal Design Criteria			
Area	**Requirement**	**Acceptance Criteria**	**Description of Compliance**
3. Design Criteria for Safety Protection Systems **f. Operating Procedures**	The DCSS must be compatible with wet or dry spent fuel loading and unloading procedures. [10 CFR 72.236(h)] Storage systems must be designed to allow ready retrieval of spent fuel for further processing or disposal. [10 CFR 72.122(l)] The DCSS must be designed to minimize the quantity of radioactive waste generated. [10 CFR 72.24(f) and 72.128(a)(5)] The applicant must describe equipment and processes proposed to maintain control of radioactive effluents. [10 CFR 72.24(l)(2)] To the extent practicable, the DCSS must be designed to facilitate decontamination. [10 CFR 72.236(I)] The applicant must establish operational restrictions to meet the limits defined in 10 CFR Part 20 and to ensure that radioactive materials in effluents and direct radiation levels associated with ISFSI operations will remain as low as is reasonably achievable (ALARA). [10 CFR 72.24(e) and 72.104(b)]	The applicant should provide potential licensees with guidance regarding the content of normal, off-normal, and accident response procedures. Cautions regarding both loading, unloading, and other important procedures should be mentioned here. Applicants may choose to provide model procedures to be used as an aid for preparing detailed site-specific procedures.	The operating procedures for the system are presented in Chapter 8 and include procedures for wet and dry loading and unloading operations. The procedures include methods for retrieving the spent fuel after storage for off-site transport or for return to the spent fuel pool. The decommissioning considerations of the system are described in Section 2.4. Operation of the system generates no radioactive waste, other than a limited amount of protective clothing and tools used during loading operations that could be easily disposed or decontaminated. The radiation protection design features of the system are presented in Section 2.3.5. Operating procedures for the system include provisions for controlling potential effluents from the system. The canister is designed to facilitate decontamination, as described in Section 2.3.5.3. Fuel assembly specifications are provided in Appendix 12A, Section 2.1.1 to ensure that doses from effluents and direct radiation are maintained ALARA.

Table 1-2 NUREG-1536 Compliance Matrix (Continued)

Chapter 2 - Principal Design Criteria			
Area	**Requirement**	**Acceptance Criteria**	**Description of Compliance**
3. Design Criteria for Safety Protection Systems **g. Acceptance Tests and Maintenance**	The DCSS design must permit testing and maintenance as required. [10 CFR 72.236(g)] SSC that are important to safety must be designed, fabricated, erected, tested, and maintained to quality standards commensurate with the importance to safety of the function to be performed. [10 CFR 72.24(c), 72.122(a), 72.122(f), and 72.128(a)(1)]	The applicant should identify the general commitments and industry codes and standards used to derive acceptance, maintenance, and periodic surveillance tests used to verify the capability of DCSS components to perform their designated functions. In addition, the applicant should discuss the methods used to assess the need for such tests with regard to specific components.	The acceptance tests and maintenance of the system are listed in Chapter 9, including the associated commitment, industry standard or NRC regulation.
3. Design Criteria for Safety Protection Systems **h. Decommissioning**	The DCSS must be compatible with wet or dry unloading facilities. [10 CFR 72.236(h)] The DCSS must be designed for decommissioning. Provisions must be made to facilitate decontamination of structures and equipment and to minimize the quantity of radioactive wastes, contaminated equipment, and contaminated materials at the time the ISFSI is permanently decommissioned. [10 CFR 72.24(f), 72.130, and 72.236(I)] The applicant must provide information concerning the proposed practices and procedures for decontaminating the site and facilities and for disposing of residual radioactive materials after all spent fuel has been removed. Such information must provide reasonable assurance that decontamination and decommissioning will adequately protect the health and safety of the public. [10 CFR 72.24(q) and 72.30(a)]	Casks should be designed for ease of decontamination and eventual decommissioning. The applicant should describe the features of the design that support these two activities.	Decommissioning of the system is discussed in Section 2.4.

Table 1-2 NUREG-1536 Compliance Matrix (Continued)

Chapter 3 – Structural Evaluation			
Area	**Regulatory Requirement**		**Description of Compliance**
1. Structures, Systems, and Components Important to Safety	Structures, systems, and components (SSC) important to safety must meet the regulatory requirements established in 10 CFR 72.24(c)(3) and (4), as well as 10 CFR 72.122(a), (b), and (c).		
	10 CFR 72.24(c)(3)	Contents of Application: Descriptions of Components Important to Safety	Component descriptions are provided in Section 1.2. Description of the structural design is provided in Section 3.1.1.
	10 CFR 72.24(c)(4)	Contents of Application: Applicable Codes and Standards	The applicable codes and standards are specified in Sections 3.1.1 and 3.1.2.
	10 CFR 72.122(a)	Overall Requirements: Quality Standards	The quality standards of the system are provided in Table 2.3-1.
	10 CFR 72.122(b)	Overall Requirements: Protection Against Environmental Conditions and natural Phenomena	The system is evaluated structurally for normal operating loads in Sections 3.4.4 and 3.4.5. Off-normal and accident loads are evaluated in Sections 11.1 and 11.2, respectively.
	10 CFR 72.122(c)	Overall Requirements: Protection Against Fires and Explosions	The system is evaluated for fire and explosive loadings in Section 11.2.

Table 1-2 NUREG-1536 Compliance Matrix (Continued)

Chapter 3 – Structural Evaluation			
Area	**Regulatory Requirement**		**Description of Compliance**
2. Radiation, Shielding, Confinement, and Subcriticality	Radiation shielding, confinement, and subcriticality must meet the regulatory requirements defined in 10 CFR 72.24(d); 10 CFR 72.124(a); and 10 CFR 72.236(c), (d), and (l).		
	10 CFR 72.24(d)	Contents of Application: Margins of Safety / Mitigation of Accident Consequences	The margins of safety for normal conditions are listed in Sections 3.4.4.1.5 and 3.4.4.2. Off-normal and accident condition margins of safety are presented in Sections 11.1 and 11.2, respectively. Adequate safety margins are maintained for all events, ensuring the mitigation of accident consequences, and the shielding, confinement, and criticality analyses presented in the SAR.
	10 CFR 72.124(a)	Criteria for Nuclear Criticality Safety: Design for Criticality Safety	The nuclear criticality safety design of the system is discussed in Sections 2.3.4 and 6.1.
	10 CFR 72.236(c)	Specific Requirements for Spent Fuel Storage Cask Approval: Maintain Subcritical Configuration	Subcriticality of the system is demonstrated in Section 6.4.3.
	10 CFR 72.236(d)	Specific Requirements for Spent Fuel Storage Cask Approval: Radiation Protection	Radiation protection of the system is demonstrated in Sections 6.4.3, 10.3 and 10.4.
	10 CFR 72.236(l)	Specific Requirements for Spent Fuel Storage Cask Approval: Maintain Confinement	Confinement of the spent fuel is discussed in Sections 7.2 and 7.3.
3. Removal of Spent Fuel	As stated in 10 CFR 72.122(f) and (h)(l), the storage system design must allow ready retrieval of spent fuel without posing operational safety problems.		The system is not adversely affected by normal, off-normal, or accident condition events as demonstrated in Sections 3.4.4.1.5, 3.4.4.2, 11.1 and 11.2. Operating procedures for removing spent fuel from the system are presented in Sections 8.2 and 8.3.
4. Design Basis Earthquake	As stated in 10 CFR 72.102(f), the design-basis earthquake (DBE) must be equal to or greater than the safe-shutdown earthquake (SSE) of nuclear plant sites previously evaluated under 10 CFR Part 100 or, in the case of sites licensed before the implementation of 10 CFR Part 100, developed under Topic III-2 of the Systematic Evaluation Program (SEP).		As described in Section 2.2.3.1, the system is designed for a seismic event that is greater than regulatory requirements.
5. Minimum Lifetime	As stated in 10 CFR 72.24(c) and 10 CFR 72.236(g), the analysis and evaluation of the structural design and performance must demonstrate that the cask system will allow storage of spent fuel for a minimum of 20 years with an adequate margin of safety.		Section 1.1 and Table 2-1 specify a 50-year design life for the system.

Table 1-2 NUREG-1536 Compliance Matrix (Continued)

Chapter 3 – Structural Evaluation		
Area	**Regulatory Requirement**	**Description of Compliance**
6. Reinforced Concrete Structures	Reinforced concrete structures may have a role in shielding, form ventilation passages and weather enclosures, and providing protection against natural phenomena and accidents. The pertinent regulations include 10 CFR 72.24(c) and 10 CFR 72.182(b) and (c).	
	10 CFR 72.24(c) Contents of Application: Design Criteria, Design Bases, Component Descriptions, Codes and Standards	A general description of the Vertical Concrete Cask (VCC) is provided in Section 1.2.1.2. The design criteria for the VCC is presented in Table 2-1. The design bases considered in the structural evaluation of the VCC are presented in Section 2.2.5.1.
	10 CFR 72.182(b) Design for Physical Protection: Design Bases / Design Criteria	This requirement is applicable to the ISFSI, not the storage system.
	10 CFR 72.182(c) Design for Physical Protection: Security System Description	This requirement is applicable to the ISFSI, not the storage system.

Table 1-2 NUREG-1536 Compliance Matrix (Continued)

Chapter 3 – Structural Evaluation		
Area	**Acceptance Criteria**	**Description of Compliance**
1. Confinement Cask **a. Steel Confinement Cask**	The structural design, fabrication, and testing of the confinement system and its redundant sealing system should comply with an acceptable code or standard, such as Section III of the Boiler and Pressure Vessel Code (B&PV) promulgated by the American Society of Mechanical Engineers (ASME [The NRC has accepted use of either Subsection NB or Subsection NC of this code.]). Other design codes or standards may be acceptable depending on their application. i. The NRC staff evaluates the proposed limitations on allowable stresses and strains in the confinement cask, reinforced concrete components, system components important to safety, and other components subject to review, by comparison with those specified in applicable codes and standards. Where certain proposed load combinations will exceed the accepted limits for localized points on the structure, the applicant should provide adequate justification to show that the deviation will not affect the functional integrity of the structure. ii. The NRC has accepted the use of applicable subsections of the ASME B&PV Code, Division 1, for components used within the confinement cask but not integrated with it. This includes the "basket" structure used in casks to restrain and position multiple fuel elements.	As specified in Section 3.1.2, the canister and basket structure are designed in accordance with the ASME Code, Section III, Division I, 1995 Edition. The canister is designed using Subsection NB from the code, while the basket structure is designed using Subsection NG criteria. A list of exceptions from the ASME code is provided in Appendix 12A, Table 12A4-1.

Table 1-2 NUREG-1536 Compliance Matrix (Continued)

Chapter 3 – Structural Evaluation		
Area	**Acceptance Criteria**	**Description of Compliance**
b. Concrete Containments	i. ACI 359 (also designated as Section III, Division 2, of the ASME B&PV Code, Subsection CC) constitutes an acceptable standard for prestressed and reinforced concrete that is an integral component of a radioactive material containment vessel that must withstand internal pressure in operation or testing. ii. If ACI 359 pertains to a given ISFSI structure, it applies to all aspects of the design, material selection, fabrication, and construction of that structure. The NRC has not accepted the proposed substitution of elements from ACI 318 or ACI 349 for any portion of ACI 359 with regard to the structure of an ISFSI. ISFSI structures to which ACI 359 applies shall also meet the minimum functional requirements of ANSI/ANS-57.9 for subject areas not specifically addressed in ACI 359.	The NAC-MPC system does not utilize concrete containment vessels. Thus, ACI-359 is not applicable.
2. Reinforced Concrete (RC) Structures Important to Safety, but not within the Scope of ACI 359	The NRC accepts the use of ACI 349 for the design, material selection and specification, and construction of all reinforced concrete structures that are not addressed within the scope of ACI 359. However, in such instances, the design, material selection and specification, and construction must also meet any additional or more stringent requirements given in ANSI/ANS-57.9, as incorporated by reference in NRC Regulatory Guide (RG) 3.60. Section V of this chapter provides additional guidance regarding specific review procedures.	As stated in Section 3.1.2, the VCC is designed in accordance with ACI-349 and ANSI/ANS-57.9.
3. Other Reinforced Concrete Structures Subject to Approval	The NRC accepts the use of either ACI 318 or ACI 349 for reinforced concrete structures that are subject to approval but are not important to safety. Section V of this chapter provides additional guidance regarding specific review procedures.	The NAC-MPC system has no concrete structures other than that addressed in #2 above.

Table 1-2 NUREG-1536 Compliance Matrix (Continued)

Chapter 3 – Structural Evaluation		
Area	**Acceptance Criteria**	**Description of Compliance**
4. Other System Components Important to Safety	The NRC accepts the use of ANSI/ANS-57.9 (together with the codes and standards cited therein) as the basic reference for ISFSI structures important to safety that are not designed in accordance with the Section III of the ASME B&PV Code. However, both the lifting equipment design and the devices for lifting system components that are important to safety must comply with American National Standards Institute (ANSI) Standard N14.6. The NRC accepts the load combinations shown in Table 3-1 for structures not designed under either Section III of the ASME B&PV Code or ACI 359. These load combinations are based upon ANSI/ANS-57.9, with supplemental definition of terms and combinations. The principal codes and standards include the following references that may apply to steel structures and components: a. American Institute of Steel Construction (AISC), "Specification for Structural Steel Buildings — Allowable Stress Design and Plastic Design" b. AISC, "Load and Resistance Factor Design Specification for Structural Steel Buildings" c. American Welding Society, "Structural Welding Code Steel," AWS D1.1 d. American Society of Civil Engineers, "Minimum Design Loads for Buildings and Other Structures," ASCE 7 9however, note that load combinations established on the basis of ANSI/ANS-57.9 [DCSS SRP Table 3-1] are to be used.). e. ACI 349-85, Appendix B, for embedments or 10.14 for composite compression sections, as applicable, when constructed of structural steel embedded in reinforced concrete.	The lifting devices of the NAC-MPC system are evaluated in accordance with NUREG-0612 and ANSI N14.6, as specified in Section 3.1.2.

Table 1-2 NUREG-1536 Compliance Matrix (Continued)

Chapter 3 – Structural Evaluation		
Area	**Acceptance Criteria**	**Description of Compliance**
5. Other Components Subject to NRC Approval	For structural design and construction of other components subject to NRC approval, the principal codes and standards include the following: a. ASCE 7 b. Uniform Building Code (UBC) c. AISC, "Specification for Structural Steel Buildings—Allowable Stress Design and Plastic Design" d. AISC "Code of Standard Practice for Steel Buildings and Bridges" e. ASME B&PV Code, Section VIII	Not applicable. All components of the system subject to NRC approval are covered by the acceptance criteria specified in the previous sections.

Table 1-2 NUREG-1536 Compliance Matrix (Continued)

Chapter 4 – Thermal Evaluation		
Area	**Regulatory Requirement**	**Description of Compliance**
1. Minimum Lifetime	10 CFR Part 72 requires an analysis and evaluation of DCSS thermal design and performance to demonstrate that the cask will permit safe storage of the spent fuel for a minimum of 20 years.	Section 1.1 and Table 2-1 specify a 50-year design life for the system. Tables 4.1-3 and 4.1-4 demonstrate that the concrete temperatures are maintained within their allowable limits.
2. Spent Fuel Cladding Protection	The spent fuel cladding must be protected against degradation that may lead to gross ruptures.	Tables 4.1-3 and 4.1-4 demonstrate that the fuel cladding temperatures are maintained within allowable limits.
3. Thermal Structures, Systems, and Components	Thermal structures, systems, and components important to safety must be described in sufficient detail to permit evaluation of their effectiveness. Applicable thermal requirements are identified, in part, in 10 CFR 72.24(c)(3), 72.24(d), 72.122(h)(1), 72.122(l), 72.128(a)(4), 72.236(f), 72.236(g), and 72.236(h).	The discussion of the thermal design features of the system is presented in Section 4.1.
	10 CFR 72.24(c)(3) Contents of Application: Descriptions of Components Important to Safety 10 CFR 72.24(d) Contents of Application: Margins of Safety / Mitigation of Accident Consequences	Tables 4.1-3 and 4.1-4 demonstrate that the temperatures are maintained within allowable limits for all components of the system, including the fuel cladding. Therefore, the system is not adversely affected by normal, off-normal, or accident condition events.
	10 CFR 72.122(h)(1) Overall Requirements: Confinement Barriers and Systems 10 CFR 72.122(l) Overall Requirements: Retrievability	The temperatures of the system are maintained within allowable limits, and do not preclude retrieval of spent fuel from the system.
	10 CFR 72.128(a)(4) Criteria for Spent Fuel Storage and Handling: Testable Heat Removal Capacity	As specified in Sections 9.1.7 and Appendix 12A, Section 3.1.7, the air temperature at the outlet vents is measured to ensure proper operation of the passive heat removal system.
	10 CFR 72.236(f) Specific Requirements for Spent Fuel Storage Cask Approval: Passive Heat Removal 10 CFR 72.236(g) Specific Requirements for Spent Fuel Storage Cask Approval: Minimum 20-year Lifetime	Section 1.1 and Table 2-1 specify a 50-year design life for the system. Tables 4.1-3 and 4.1-4 demonstrate that the concrete temperatures are maintained within their allowable limits.
	10 CFR 72.236(h) Specific Requirements for Spent Fuel Storage Cask Approval: Wet/Dry Loading and Unloading Compatibility	The operating procedures for the system presented in Chapter 8 include procedures for wet and dry loading and unloading operations.

Table 1-2 NUREG-1536 Compliance Matrix (Continued)

Chapter 4 – Thermal Evaluation		
Area	**Acceptance Criteria**	**Description of Compliance**
1. Long-term Cladding Temperatures	Fuel cladding (zircaloy) temperature at the beginning of dry cask storage should generally be below the anticipated damage-threshold temperatures for normal conditions and a minimum of 20 years of cask storage (Refs. 13 and 14). Ref 13: UCID-21181, "Spent Fuel Cladding Integrity During Dry Storage" Ref 14: PNL-6189, "Recommended Temperature Limits for Dry Storage of Spent Light-Water Zircalloy clad fuel Rods in Inert Gas"	As shown in Table 4.1-4, the fuel cladding temperatures are maintained below 806°F (430°C) for both Zircaloy-clad and Stainless Steel-Clad fuel. This temperature is within the recommended temperature limits for Zircaloy-clad fuel (PNL-6189) and within the limits for Stainless Steel-clad fuel (EPRI TR-106440) for normal, off-normal and accident conditions.
2. Short Term Cladding Temperatures	Fuel cladding temperature should generally be maintained below 570 °C (1058 °F) for short-term accident conditions, short-term off-normal conditions, and fuel transfer operations (e.g., vacuum drying of the cask or dry transfer). (PNL-4835)	As shown in Table 4.1-4, the fuel cladding temperature for both Zircaloy and Stainless Steel are maintained below 806°F (430°C) for short term off-normal or accident condition events.
3. Maximum Internal Pressure	The maximum internal pressure of the cask should remain within its design pressures for normal, off-normal, and accident conditions assuming rupture of 1 percent, 10 percent, and 100 percent of the fuel rods, respectively. Assumptions for pressure calculations include release of 100 percent of the fill gas and 30 percent of the significant radioactive gases in the fuel rods.	The normal condition pressure calculation is presented in Section 4.4.5. The accident condition pressure calculation is presented in Section 11.2.1. The off-normal condition is bounded by the accident condition, which assumes 100% failure of the cladding.
4. Maximum Material Temperatures	Cask and fuel materials should be maintained within their minimum and maximum temperature criteria for normal, off-normal, and accident conditions in order to enable components to perform their intended safety functions.	Tables 4.1-3 and 4.1-4 demonstrate that the temperatures are maintained within allowable limits for all components of the system, including the fuel cladding. Therefore, the system is not adversely affected by normal, off-normal, or accident condition events.
5. Fuel Cladding Protection	For each fuel type proposed for storage, the DCSS should ensure a very low probability (e.g., 0.5 percent per fuel rod) of cladding breach during long-term storage.	As concluded in PNL-6189 (Zircaloy) and EPRI TR-106449 (Stainless Steel), the probability of cladding breech is very low when the cladding temperature is maintained below allowable limits.
6. Long-Term Cladding Damage	Fuel cladding damage resulting from creep cavitation should be limited to 15 percent of the original cladding cross-sectional area during dry storage. (UCID-21181)	The maximum fuel cladding temperatures are determined in accordance with PNL-6189 (Zircaloy) and EPRI TR-106440 (Stainless Steel). Calculations were not performed using the methodology of UCID-21181.
7. Passive Cooling	The cask system should be passively cooled. [10 CFR 72.236(f)]	As stated in Sections 1.2 and 4.1, the system is passively cooled.
8. Thermal Operating Limits	The thermal performance of the cask should be within the allowable design criteria specified in SAR Section 2 (e.g., materials, decay heat specifications) and SAR Section 3 (e.g., thermal stress analysis) for normal, off-normal, and accident conditions.	The thermal stress analyses of the canister and VCC for normal conditions are provided in Sections 3.4.4.1.1 and 3.4.4.2.3, respectively. The canister is evaluated for off-normal thermal stresses in Section 11.1.4.2, and the VCC is analyzed for accident thermal stresses in Section 11.2.10.

Table 1-2 NUREG-1536 Compliance Matrix (Continued)

Chapter 5 – Shielding Evaluation		
Area	**Regulatory Requirement**	**Description of Compliance**
1. Shielding System Description	10 CFR Part 72 requires that spent fuel radioactive waste storage and handling systems be designed with suitable shielding to provide adequate radiation protection under both normal and accident conditions. Consequently, the DCSS application must describe the shielding structures, systems, and components (SSCs) important to safety in sufficient detail to allow the NRC staff to thoroughly evaluate their effectiveness. It is the responsibility of the vendor, the facility owner, and the NRC staff to analyze such SSCs with the objective of assessing the impact of direct radiation doses on public health and safety.	A general description of the system is provided in Section 1.2. A detailed description of the shielding features of the system are provided in Section 5.1.

Table 1-2 NUREG-1536 Compliance Matrix (Continued)

Chapter 5 – Shielding Evaluation			
Area	**Regulatory Requirement**		**Description of Compliance**
2. Protection During Accidents	In addition, SSCs important to safety must be designed to withstand the effects of both credible accidents and severe natural phenomena without impairing their capability to perform their safety functions. The applicable shielding requirements are identified, in part, in 10 CFR 72.24(c)(3), 72.24(d), 72.104(a), 72.106(b), 72.122(b), 72.122(c), 72.128(a)(2), and 72.236(d).		
	10 CFR 72.24(c)(3)	Contents of Application: Descriptions of Components Important to Safety	A description of the shielding components of the system is provided in Section 5.1.
	10CFR 72.24(d)	Contents of Application: Margins of Safety / Mitigation of Accident Consequences	The design basis dose rates for accident conditions are listed in Section 10.2.2. Specific details of the dose rates due to the tip-over accident are presented in Section 11.2.12.3.
	10 CFR 72.104(a)	Criteria for Radioactive Materials in Effluents and Direct Radiation from an ISFSI or MRS: Annual Site Boundary Dose Limit	The controlled area boundary dose calculations and minimum site boundary distances are presented in Section 10.4.
	10 CFR 72.106(b)	Controlled Area of an ISFSI or MRS: Design Basis Accident Site Boundary Dose Limit	The accident condition dose rates are discussed in Section 10.2.2.
	10 CFR, 72.122(b)	Overall Requirements: Protection Against Environmental Conditions and Natural Phenomena	Evaluation of the system for off-normal and accident condition events is provided in Sections 11.1 and 11.2. The radiological consequences of each event are addressed.
	10 CFR 72.122(c)	Overall Requirements: Protection Against Fires and Explosions	The radiological consequences of a fire accident are provided in Section 11.2.5.3. The radiological consequences of an explosion are provided in Section 11.2.3.3.
	10 CFR 72.128(a)(2)	Criteria for Spent Fuel Storage and Handling: Radiation Protection	The dose rate results demonstrating the radiation protection of the system are presented in Section 5.1.
	10 CFR 72.236(d)	Specific Requirements for Spent Fuel Storage Cask Approval: Radiation Protection	As described above, the normal condition controlled area boundary dose rates are provided in Section 10.4. The accident condition doses are discussed in Section 10.2.2.

Table 1-2 NUREG-1536 Compliance Matrix (Continued)

Chapter 5 – Shielding Evaluation		
Area	**Acceptance Criteria**	**Description of Compliance**
1. Minimum Distance from Controlled Area Boundary	The minimum distance from each spent fuel handling and storage facility to the controlled area boundary must be at least 100 meters. The "controlled area" is defined in 10 CFR 72.3 as the area immediately surrounding an ISFSI or monitored retrievable storage (MRS) facility, for which the licensee exercises authority regarding its use and within which ISFSI operations are performed.	As described in Section 10.4, the minimum allowable controlled area boundary distance is 150 meters.
2. Controlled Area Boundary Dose Limits	The cask vendor must show that, during both normal operations and anticipated occurrences, the radiation shielding features of the proposed DCSS are sufficient to meet the radiation dose requirements in Sections 72.104(a). Specifically, the vendor must demonstrate this capability for a typical array of casks in the most bounding site configuration. For example, the most bounding configuration might be located at the minimum distance (100 meters) to the controlled area boundary, without any shielding from other structures or topography.	Section 10.4 presents the controlled area boundary dose rate evaluation for a typical array configuration. The minimum allowable controlled area boundary distance is 150 meters without taking credit for shielding provided by any intermediate structures or topography.
3. ALARA	Dose rates from the cask must be consistent with a well-established "as low as reasonably achievable" (ALARA) program for activities in and around the storage site.	The dose rates for the system are presented in Section 5.1. These dose rates are within the allowables specified in Section 10.2.1, which are consistent with ALARA principles.
4. Maximum Accident Controlled Area Boundary Dose	After a design-basis accident, an individual at the boundary or outside the controlled area shall not receive a dose greater than 5 rem to the whole body or any organ.	Section 10.2.2 indicates that the controlled area boundary dose as a result of an accident will not exceed 5 rem.
5. Occupational Dose Limits	The proposed shielding features must ensure that the DCSS meets the regulatory requirements for occupational and radiation dose limits for individual members of the public, as prescribed in 10 CFR Part 20, Subparts C and D.	Occupational doses for typical loading operations are provided in Section 10.3. In practice, occupational doses would be controlled on a site-specific basis by the operator of the ISFSI.

Table 1-2 NUREG-1536 Compliance Matrix (Continued)

Chapter 6 – Criticality Evaluation			
Area	**Regulatory Requirement**		**Description of Compliance**
	Spent fuel storage systems must be designed to remain subcritical unless at least two unlikely independent events occur. Moreover, the spent fuel cask must be designed to remain subcritical under all credible conditions. Regulations specific to nuclear criticality safety of the cask system are specified in 10 CFR 72.124 and 72.236(c). Other pertinent regulations include 10 CFR 72.24(c)(3), 72.24(d), and 72.236(g). Normal and accident conditions to be considered are also identified in 10 CFR Part 72.		
	10 CFR 72.24(c)(3)	Contents of Application: Descriptions of Components Important to Safety	A general description of the system is provided in Section 1.2, with a detailed description of the criticality safety features of the system provided in Section 6.1.
	10 CFR 72.24(d)	Contents of Application: Margins of Safety / Mitigation of Accident Consequences	Section 6.4 presents the results of the criticality evaluation of the transfer cask and storage cask.
	10 CFR 72.124	Criteria for Nuclear Criticality Safety	The criteria for criticality safety are provided in Sections 2.3.4 and 6.1.
	10 CFR 72.236(c)	Specific Requirements for Spent Fuel Storage Cask Approval: Maintain Subcritical Configuration	Section 6.4 presents the results of the criticality evaluation of the storage cask for the most reactive credible conditions.
	10 CFR 72.236(g)	Specific Requirements for Spent Fuel Storage Cask Approval: Minimum 20-year Lifetime	Section 1.1 and Table 2-1 specify a 50-year design life for the system.

Table 1-2 NUREG-1536 Compliance Matrix (Continued)

Chapter 6 – Criticality Evaluation		
Area	**Acceptance Criteria**	**Description of Compliance**
1. Subcriticality Margin	The multiplication factor (k_{eff}), including all biases and uncertainties at a 95-percent confidence level, should not exceed 0.95 under all credible normal, off-normal, and accident conditions.	As stated in Section 6.1, the maximum allowable multiplication factor (k_s) for the system is 0.95, including adjustment for all biases and uncertainties, as calculated in Section 6.5.
2. Double Contingency	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.	As stated in Section 6.1, the criticality analyses are performed for the most reactive credible configuration of the cask, at the highest enrichment, without credit for fuel burnup, and at the most reactive internal water moderator density, even though it is stated that water intrusion is not a credible event. Therefore, criticality cannot occur unless two separate events, such as (1) misloading a higher than design-basis enrichment, unirradiated fuel assembly and (2) water intrusion, occur.
3. Criticality Design Features	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.	As stated in Section 6.1, the criticality safety of the design is based on geometry and fixed neutron poisons. Recently proposed rule changes (Federal Register, June 9, 1998) include discussion clarifying the 10 CFR 72.124(b) requirement to verify the "continued efficacy" of neutron poisons as applicable only to wet storage systems, and not to dry, provided that the effectiveness of the poisons is demonstrated at the outset.
4. Conservative Assumptions	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 percent for fixed neutron absorbers when subject to standard acceptance tests.	Section 6.1 provides a list of conservative assumptions that are used in the criticality safety evaluation. No fuel burnup is assumed, and only 75% of the minimum ^{10}B loading on the Boral plates is used. Also, no integral fuel burnable neutron absorbers, nor fission product neutron poisons, are considered in the analysis.

Table 1-2 NUREG-1536 Compliance Matrix (Continued)

Chapter 7 – Confinement Evaluation		
Area	**Regulatory Requirement**	**Description of Compliance**
1. Description of Structures, Systems, and Components Important to Safety	The SAR must describe the confinement structures, systems, and components (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)]	A general description of the system is provided in Section 1.2, with a detailed description of the confinement features of the system provided in Section 7.1.
2. Protection of Spent Fuel Cladding	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)]	As described in Sections 7.2.1 and 7.3, the integrity of the canister is maintained under normal and accident conditions. Therefore, the inert helium atmosphere is maintained in the canister, protecting the fuel cladding against degradation.
3. Redundant Sealing	The cask design must provide redundant sealing of the confinement boundary. [10 CFR 72.236(e)]	As described in Section 7.1.3.2, the canister is sealed after loading by means of a redundant lid system.
4. Monitoring of Confinement System	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)]	As described in Appendix 12A, Section 3.1.7, the system is designed for air outlet temperature monitoring. The canister is designed to withstand all normal, off-normal, and accident condition events, while maintaining the integrity of the confinement boundary.
5. Instrumentation	The design must provide instrumentation and controls to monitor systems that are important to safety over anticipated ranges for normal and off-normal operation. In addition, the applicant must identify those control systems that must remain operational under accident conditions. [10 CFR 72.122(i)]	As described in Appendix 12A, Section 3.1.7, the inlet and outlet vent air temperatures are monitored to ensure that temperatures within the confinement boundary are maintained within allowable limits.
6. Release of Nuclides to the Environment	The applicant must estimate the quantity of radionuclides expected to be released annually to the environment. [10 CFR 72.24(l)(1)]	As described in Sections 7.2.1 and 7.3, the leaktight integrity of the confinement boundary is maintained during all postulated normal and accident condition events. Therefore, there is no release of radionuclides to the environment.

Table 1-2 NUREG-1536 Compliance Matrix (Continued)

Chapter 7 – Confinement Evaluation		
Area	**Regulatory Requirement**	**Description of Compliance**
7. Evaluation of Confinement System	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 72.236(l) and 10 CFR 72.24(d)] In addition, 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)]	The confinement system is analyzed for normal conditions in Sections 3.4.3.2 and 3.4.4.1, and for off-normal, and accident conditions in Sections 11.1 and 11.2, respectively.
8. Annual Dose Limit in Effluents and Direct Radiation from an Independent Spent Fuel Storage Installation (ISFSI)	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)]	The site boundary dose calculations and minimum site boundary distances are presented in Section 10.4.

Table 1-2 NUREG-1536 Compliance Matrix (Continued)

Chapter 7 – Confinement Evaluation		
Area	**Acceptance Criteria**	**Description of Compliance**
1. Redundant Sealing	The cask design must provide redundant sealing of the confinement boundary sealing surface. Typically, this means that field closures of the confinement boundary must either have double seal welds or double metallic o-ring seals.	As described in Section 7.1.3.2, the canister is sealed after loading by means of a redundant lid system.
2. Code Compliance	The confinement design must be consistent with the regulatory requirements, as well as the applicant's "General Design Criteria" reviewed in Chapter 2 of this SRP. The NRC staff has accepted construction of the primary confinement barrier in conformance with Section III, Subsections NB or NC, of the Boiler and Pressure Vessel (B&PV) Code promulgated by the American Society of Mechanical Engineers (ASME [This code defines the standards for all aspects of construction, including materials, design, fabrication, examination, testing, inspection, and certification required in the manufacture and installation of components.]). In such instances, the staff has relied upon Section III to define the minimum acceptable margin of safety; therefore, the applicant must fully document and completely justify any deviations from the specifications of Section III. In some cases after careful and deliberate consideration, the staff has made exceptions to this requirement.	The codes and standards utilized for the confinement system design are specified in Section 7.1.1. ASME Section III, Subsection NB is utilized for the design of the canister.
3. Maximum Allowable Leakage Rates	The applicant must specify the maximum allowed leakage rates for the total primary confinement boundary and redundant seals (Applicants frequently display this information in tabular form, including the leakage rate of each seal.). In addition, the applicant's leakage analysis should be consistent with the principles specified in the "American National Standard for Leakage Tests on Packages for Shipment of Radioactive Materials" (ANSI N14.5). Generally, the allowable leakage rate must be evaluated for its radiological consequences and its effect on maintaining the necessary inert atmosphere within the cask.	As specified in Sections 7.2.1 and 7.3, leakage from the confinement system under normal, off-normal, and accident conditions is not credible because the canister is demonstrated to be leaktight.

Table 1-2 NUREG-1536 Compliance Matrix (Continued)

Chapter 7 – Confinement Evaluation		
Area	**Acceptance Criteria**	**Description of Compliance**
4. Monitoring and Surveillance	The applicant should describe the proposed monitoring capability and/or surveillance plans for mechanical closure seals. In instances involving welded closures, the staff has previously accepted that no closure monitoring system is required. This practice is consistent with the fact that other welded joints in the confinement system are not monitored. However, the lack of a closure monitoring system has typically been coupled with a periodic surveillance program that would enable the licensee to take timely and appropriate corrective actions to maintain safe storage conditions after closure degradation. The discussion in (a) below taken from Chapter 2 of this SRP expands on the requirement for continuous monitoring. (a) Continuous Monitoring The Office of the General Counsel (OGC) has developed an opinion as to what constitutes "continuous monitoring" as required in 10 CFR Part 72.122(h)(4). The staff, in accordance with that opinion has concluded that both routine surveillance programs and active instrumentation meets the intent of "continuous monitoring." Cask vendors may propose, as part of the SAR, either active instrumentation and/or surveillance to show compliance with 10 CFR Part 72.122(h)(4). The reviewer should note that some DCSS designs may contain a component or feature whose continued performance over the licensing period has not been demonstrated to staff with a sufficient level of confidence. Therefore the staff may determine that active monitoring instrumentation is required to provide for the detection of component degradation or failure. This particularly applies to components whose failure immediately affects or threatens public health and safety. In some cases the vendor or staff in order to demonstrate compliance with 10 CFR Part 72.122(h)(4), may propose a technical specification requiring such instrumentation as part of the initial use of a cask system. After initial use, and if warranted and approved by staff, such instrumentation may be discontinued or modified.	The system utilizes welded closures, as specified in Section 7.1. Therefore, no monitoring system is required. Routine surveillance is required as specified in Appendix 12A of Chapter 12.

Table 1-2 NUREG-1536 Compliance Matrix (Continued)

Chapter 7 – Confinement Evaluation		
Area	**Acceptance Criteria**	**Description of Compliance**
5. Non-Reactive Environment	The cask must provide a non-reactive environment to protect fuel assemblies against fuel cladding degradation, which might otherwise lead to gross rupture. Measures for providing a non-reactive environment within the confinement cask typically include drying, evacuating air and water vapor, and backfilling with a non-reactive cover gas (such as helium). For dry storage conditions, experimental data have not demonstrated an acceptably low oxidation rate for UO_2 spent fuel, over the 20-year licensing period, to permit safe storage in an air atmosphere. Therefore, to reduce the potential for fuel oxidation and subsequent cladding failure, an inert atmosphere (e.g., helium cover gas) has been used for storing UO_2 spent fuel in a dry environment (See Chapter 8 of this SRP for more detailed information on the cover gas filling process.). Note that other fuel types, such as graphite fuels for the high-temperature gas-cooled reactors (HTGRs), may not exhibit the same oxidation reactions as UO_2 fuels and, therefore, may not require an inert atmosphere. Applicants proposing to use atmospheres other than inert gas should discuss how the fuel and cladding will be protected from oxidation.	As described in Sections 7.1.1.1 and 7.1.2, the confinement system is vacuum dried and backfilled with inert helium gas during loading operations.

Table 1-2 NUREG-1536 Compliance Matrix (Continued)

Chapter 8 – Operating Procedures		
Area	**Regulatory Requirement**	**Description of Compliance**
Health and Safety	1. The applicant must develop operating procedures that adequately protect health and minimize danger to life or property. [10 CFR 72.40(a)(5)]	Operating procedures are provided in Chapter 8. Notes and Cautions are listed among the steps provided to emphasize steps important to maintaining health and safety.
ALARA	2. The applicant must establish operational restrictions to meet the regulatory requirements of 10 CFR Part 20 and objective limits that are as low as is reasonably achievable (ALARA) for radioactive materials in effluents and direct radiation levels associated with ISFSI operations. [10 CFR 72.104(b) and 10 CFR 72.24(e)]	Section 8.0 specifies that the procedures are developed to maintain occupational dose ALARA. Automated welding systems and temporary shielding are utilized to minimize worker dose during canister loading operations. Appendix 12A, Section 3.2.1 specifies maximum external dose rates to maintain reasonable dose level within a cask array for routine surveillance and inspection activities.
Control of Radioactive Effluents	3. The applicant must describe all equipment and processes used to maintain control of radioactive effluents. [10 CFR 72.24(l)(2)]	As described in Sections 7.1.1.4, 7.2.1, and 7.3, there are no radioactive effluents in routine operations other than pool water and helium gas that are removed from the canister. These effluents are routinely handled in Licensee operations.
Written Procedures	4. The general licensee shall conduct activities related to storage of spent fuel in accordance with written procedures. [10 CFR 72.212(b)(9)] 5. Vendors seeking approval of a cask design shall ensure that written procedures and appropriate tests are established before initial use of the casks. In addition, the vendor must provide a copy of these procedures and tests to each prospective cask user. [10 CFR 72.234(f)]	Written procedures for the system are provided in Chapter 8. These procedures are intended to provide general operational guidance for use of the system. These procedures would be used by an ISFSI operator to develop detailed, site specific procedures for use of the system.
Wet or Dry Loading and Unloading Facilities	6. The cask must be compatible with wet or dry spent fuel loading and unloading facilities. [10 CFR 72.236(h)]	The operating procedures provided in Chapter 8 include procedures for wet and dry loading and unloading operations.
Decontamination Features	7. To the extent practicable, the design of the cask must facilitate decontamination. [10 CFR 72.236(i)]	The canister is designed to facilitate decontamination as described in Section 2.3.5.3. As described in Section 8.1.1, the annulus between the canister and transfer cask is filled with clean water prior to placement in the fuel pool to minimize the potential for contamination of the surface of the canister.

Table 1-2 NUREG-1536 Compliance Matrix (Continued)

Chapter 8 – Operating Procedures		
Area	**Regulatory Requirement**	**Description of Compliance**
Ready Retrieval of Spent Fuel	8. The design of storage systems must allow ready retrieval of spent fuel for further processing or disposal. [10 CFR 72.122(l)]	The procedure provided in Sections 8.2 and 8.3 specify the steps necessary for retrieval of the spent fuel from the system for further processing or disposal.
Radioactive Waste Generation	9. The design of the cask must minimize the quantity of radioactive waste generated. [10 CFR 72.128(a)(5) and 10 CFR 72.24(f)]	Operation of the system generates no radioactive waste, other than a limited amount of protective clothing and tools used during loading operations that could be easily disposed or decontaminated.
Inspection, Maintenance, and Testing	10. The design of structures, systems, and components (SSCs) that are important to safety must permit inspection, maintenance, and testing. [10 CFR 72.122(f)]	Section 9.2 specifies the inspection and maintenance activities required for the system.

Table 1-2 NUREG-1536 Compliance Matrix (Continued)

Chapter 8 – Operating Procedures		
Area	**Acceptance Criteria**	**Description of Compliance**
Scope of Application	1. Major operating procedures apply to the principal activities expected to occur during dry cask storage. The expected scope of activities for the SAR operating procedure descriptions is described in Section II, "Areas of Review" (*of the SRP*), as well as Section 8 of Regulatory Guide 3.61. Operating procedure descriptions should be submitted to address the cask design features and planned operations.	The operating procedures provided in Chapter 8 cover all planned operations of the system, including loading of spent fuel, placement of the system at the site, and unloading of the system.
Process Control and Hazard Mitigation	2. Operating procedure descriptions should identify measures to control processes and mitigate potential hazards that may be present during planned normal operations. Section V, "Review Procedures" (*of the SRP*), discusses previously identified processes and potential hazards.	The operating procedures provided in Chapter 8 include Notes and Cautions to indicate steps important in maintaining safety.
Operating Controls and Limits	3. Operating procedure descriptions should ensure conformance with the applicable operating controls and limits described in the technical specifications provided in SAR Section 12.	The operating controls and limits specified in Section 12.1 are included with the appropriate procedures in Chapter 8 and the Technical Specifications in Appendix 12A. of Chapter 12.

Table 1-2 NUREG-1536 Compliance Matrix (Continued)

Chapter 8 – Operating Procedures		
Area	**Acceptance Criteria**	**Description of Compliance**
Operational Planning	4. Operating procedure descriptions should reflect planning to ensure that operations will fulfill the following acceptance criteria:	
	a. Occupational radiation exposures will remain ALARA	As stated in Section 8.0, the operating procedures are developed to support maintaining occupational doses ALARA.
	b. Effective measures will be taken to preclude potential unplanned and uncontrolled releases of radioactive materials	Sections 8.1.1 and 8.3 include steps to preclude releases of radioactive material during loading and unloading operations. As stated in Sections 7.2.1 and 7.3, release of radioactive material from the system under normal, off-normal and accident conditions is not a credible event.
	c. Offsite dose rates will be maintained within the limits of 10 CFR Part 20 and 10 CFR 72.104 for normal operations, and 10 CFR 72.106 for accident conditions.	Section 10.4 presents the site boundary dose rate evaluation, including the minimum controlled area boundary distance needed to meet an annual dose limit of 25 mrem for normal conditions. Section 10.2.2 indicates that the accident condition controlled area boundary dose will not exceed 5 rem to any organ.
	In addition, the operating procedure descriptions should support and be consistent with the bases used to estimate radiation exposures and total doses (Refer to Chapter 10 of this SRP).	The operating procedures specified in Chapter 8 and the previous cask loading and unloading experience of NAC support the calculation of occupational dose rates presented in Section 10.3.

Table 1-2 NUREG-1536 Compliance Matrix (Continued)

Chapter 8 – Operating Procedures		
Area	**Acceptance Criteria**	**Description of Compliance**
Surveillance, Maintenance, and Contingency Plans	5. Operating procedure descriptions should include provisions for the following activities:	
	a. testing, surveillance, and monitoring of the stored material and casks during storage and loading and unloading operations	Section 9.2 specifies the inspection and maintenance activities required for the system during storage. The limits established in Appendix 12A, Sections 2.0 and 3.0, are provided to ensure that the spent fuel is protected during loading and unloading operations.
	b. maintenance of casks and cask functions during storage	Normal operational maintenance and surveillance activities are specified in Section 9.2.
	c. contingency actions triggered by inspections, checks, observations, instrument readings, and so forth (Some of these may involve off-normal conditions addressed in SAR Section 11.).	These activities include contingency actions that may be required as a result of the inspection.
Cladding Protection	6. As required by 10 CFR 72.122(h)(1), the operating procedure descriptions should facilitate reducing the amount of water vapor and oxidizing material within the confinement cask to an acceptable level to protect the spent fuel cladding against degradation that might otherwise lead to gross ruptures.	As specified in Appendix 12A, Sections 3.1.2 and 3.1.3, the canister is vacuum dried to eliminate water and backfilled with inert helium gas during fuel loading operations to protect the fuel cladding against oxidation.

Table 1-2 NUREG-1536 Compliance Matrix (Continued)

Chapter 9 – Acceptance Test and Maintenance Program		
Area	**Regulatory Requirement**	**Description of Compliance**
1. Testing and Maintenance	a. The SAR must describe the applicant's program for preoperational testing and initial operations. [10 CFR 72.24(p)]	Section 9.1 presents the acceptance testing for the system.
	b. The cask design must permit maintenance as required. [10 CFR 72.236(g)]	Section 9.2 presents the maintenance activities for the system.
	c. Structures, systems, and components (SSCs) important to safety must be designed, fabricated, erected, tested, and maintained to quality standards commensurate with the importance to safety of the function they are intended to perform. [10 CFR 72.122(a), 10 CFR 72.122(f), 10 CFR 72.128(a)(1), and 10 CFR 72.24(c)]	The acceptance tests and maintenance activities presented in Sections 9.1 and 9.2 are performed to verify compliance with the design bases and criteria, and that the system continues to perform as designed.
	d. The applicant or licensee must establish a test program to ensure that all required testing is performed to meet applicable requirements and acceptance criteria. In addition, at least 30 days before the receipt of spent fuel, the licensee must submit to the NRC a report concerning the pre-operational test acceptance criteria and test results. [10 CFR 72.162 and 10 CFR 72.82(e)]	The testing and maintenance provided in Sections 9.1 and 9.2 are intended to be used by an ISFSI user in the development of site-specific programs.
	e. The applicant or licensee 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 72.236(l)]	The acceptance tests presented in Section 9.1 demonstrate that the system will maintain confinement of the spent fuel under normal, off-normal, and accident conditions.
	f. The applicant or licensee must inspect the cask to ascertain that there are no cracks, pinholes, uncontrolled voids, or other defects that could significantly reduce confinement effectiveness. [10 CFR 72.236(j)]	As described in Section 9.1.1, the canister is visually and non-destructively examined prior to use.
	g. The applicant must perform, and make provisions that permit the Commission to perform, tests that the Commission deems necessary or appropriate. [10 CFR 72.232(b)]	As described in Section 9.2.2, a thermal test is to be performed on the first in-service system. This testing is performed in accordance with a historical NRC certification requirement. Provisions shall be made, as necessary, to facilitate additional NRC imposed testing as required.
	h. The general licensee must accurately maintain the record provided by the cask supplier showing any maintenance performed on each cask. This record must include evidence that any maintenance and testing have been conducted under an NRC-approved quality assurance (QA) program. [10 CFR 72.212(b)(8)]	Records of maintenance activities would be maintained by the ISFSI user, and thus are not applicable.
	The applicant or licensee must assure that the casks are conspicuously and durably marked with a model number, unique identification number, and the empty weight [10 CFR 72.236(k)]	As specified in Section 9.1.8, each system is to be marked with the model number, unique cask number, empty weight, and additional information.

Table 1-2 NUREG-1536 Compliance Matrix (Continued)

Chapter 9 – Acceptance Test and Maintenance Program		
Area	**Regulatory Requirement**	**Description of Compliance**
2. Resolution of Issues Concerning Adequacy or Reliability	The SAR must identify all SSCs important to safety for which the applicant cannot demonstrate functional adequacy and reliability through previous acceptable evidence. For this purpose, acceptable evidence may be established in any of the following ways: • prior use for the intended purpose • reference to widely accepted engineering principles • reference to performance data in related applications In addition, the SAR should include a schedule showing how the applicant or licensee will resolve any associated safety questions before the initial receipt of spent fuel. [10 CFR 72.24(i)]	As described in Sections 3.1 and 3.3, the design of the system is based on industry standard codes and standards for materials and margins of safety. The acceptance tests specified in Section 9.1 are performed to demonstrate the adequacy of each fabricated system in accordance with applied Codes and Standards. The system does not rely on any materials or design standards that lack acceptable evidence of functional adequacy.
3. Cask Identification	The applicant or licensee must conspicuously and durably mark the cask with a model number, unique identification number, and empty weight. [10 CFR 72.236(k)]	As specified in Section 9.1.8, each system is to be marked with the model number, unique cask number, empty weight, and additional information.

Table 1-2 NUREG-1536 Compliance Matrix (Continued)

Chapter 9 – Acceptance Tests and Maintenance Program		
Area	**Acceptance Criteria**	**Description of Compliance**
Confinement System	American Society of Mechanical Engineers (ASME), "Boiler and Pressure Vessel (B&PV) Code", Section III, Subsection NB or NC "American National Standard for Radioactive Materials— Leakage Tests on Packages for Shipment" (ANSI N14.5-1987)	As specified in Section 3.1.2, the canister is designed in accordance with the ASME Code, Section III, Subsection NB. Exceptions to the Code are provided in Appendix 12A, Table 12A4-1. The confinement system is leak tested in accordance with ANSI N14.5 following shield lid welding as specified in Appendix 12A, Section 3.1.4.
Confinement Internals (e.g., basket)	ASME B&PV Code, Section III, Subsection NG	As specified in Section 3.1.2, the basket structure is designed in accordance with the ASME Code, Section III, Subsection NG.
Metal Cask Overpack	ASME B&PV Code, Section VIII	Not applicable.
Concrete Cask Overpack	American Concrete Institute (ACI) Standards 318 and 349, as appropriate	As stated in Section 3.1.2, the VCC is designed in accordance with ACI-349 and ANSI/ANS-57.9.
Other Metal Structures	ASME B&PV Code, Section III, Subsection NF American Institute of Steel Construction (AISC), "Manual of Steel Construction"	Not applicable.

Table 1-2 NUREG-1536 Compliance Matrix (Continued)

Chapter 10 – Radiation Protection		
Area	**Regulatory Requirement**	**Description of Compliance**
1. Effluent and Direct Radiation	Criteria for radioactive material released due to effluents and direct radiation from an ISFSI or MRS are contained 10 CFR 72.104. 10 CFR 72.104 Criteria for Radioactive Materials in Effluents and Direct Radiation from an ISFSI or MRS	The controlled area boundary dose calculations and minimum controlled area boundary distances are presented in Section 10.4.
2. Occupational Exposures	Criteria for Occupational Exposures are contained in 10 CFR 20.1201, 10 CFR 20.1207, 10 CFR 20.1208, and 10 CFR 20.1301 10 CFR 20.1201 Occupational Dose Limits for Adults 10 CFR 20.1207 Occupational Dose Limits for Minors 10 CFR 20.1208 Dose to an Embryo/Fetus 10 CFR 20.1301 Dose Limits for Individual Members of the Public	Occupational doses for typical loading operations are provided in Section 10.3. In practice, occupational doses would be controlled on a site-specific basis by the operator of the ISFSI.
3. Public Exposures	Criteria for public exposures under normal and accident conditions are contained within. [10 CFR 72.104 and 10 CFR 72.106] 10 CFR 72.104 Criteria for Radioactive Materials in Effluents and Direct Radiation from an ISFSI or MRS 10 CFR 72.106 Controlled Area of an ISFSI or MRS	The controlled area boundary dose calculations and minimum site boundary distances are presented in Section 10.4. Section 10.2.2 indicates that the controlled area boundary dose as a result of an accident will not exceed 5 rem to any organ, exclusive of skin.

Table 1-2 NUREG-1536 Compliance Matrix (Continued)

Chapter 10 – Radiation Protection		
Area	**Regulatory Requirement**	**Description of Compliance**
4. ALARA	Criteria for ALARA are contained within 10 CFR 20.1101, 10 CFR 72.24(e), 10 CFR 72.104(b), and 10 CFR 72.126(a)	
	10 CFR 20.1101 Radiation Protection Programs 10 CFR 72.24(e) Contents of Application: ALARA Features	The description of the radiation protection and ALARA considerations of the system are provided in Section 10.1.
	10 CFR 72.104(b) Criteria for Radioactive Materials in Effluents and Direct Radiation from an ISFSI or MRS: Operational Restrictions	The design basis for radiation protection is presented in Section 10.2.
	10 CFR 72.126(a) Criteria for Radiological Protection: Exposure Control	Operational methods utilized to provide radiation protection are discussed in Section 10.1.3.

Table 1-2 NUREG-1536 Compliance Matrix (Continued)

Chapter 10 – Radiation Protection		
Area	**Acceptance Criteria**	**Description of Compliance**
1. Design Criteria	Limitations on dose rates associated with direct radiation from the cask are established on the basis of the shielding and confinement evaluations in order to satisfy the regulatory requirements for public dose limits. As stated in 10 CFR Part 72.104, during normal operations and anticipated occurrences, the annual dose equivalent to a real individual located beyond the controlled area, must not exceed the limits discussed below.	The dose rate design criteria are specified in Section 10.2.1.
2. Occupational Exposures	a. dose limits for adults: 5 rem/yr (total effective dose equivalent) b. dose limits for minors: 0.5 rem/yr c. dose to an embryo or fetus (declared pregnant woman): 0.5 rem during entire pregnancy	Occupational doses for typical loading operations are provided in Section 10.3. In practice, occupational doses would be controlled on a site-specific basis by the operator of the ISFSI.
3. Public Exposures	**a. Normal Conditions:** whole body: 25 mrem/yr thyroid: 75 mrem/yr other organ: 25 mrem/yr These doses include the cumulative effects of other nuclear fuel cycle facilities that may be at the same location as the storage system (i.e., the nuclear power plant) and apply to the limiting real individual of the general public residing at a permanent location nearest the facility. **b. Accident Conditions and Natural Phenomenon Events** 5 rem to the whole body or any organ of any individual located at or beyond the nearest boundary of the controlled area.	The controlled area boundary dose calculations and minimum controlled area boundary distances under normal conditions are presented in Section 10.4. Contribution to the controlled area boundary dose rate from other facilities co-located with the ISFSI are beyond the scope of the SAR, and are addressed on a site-specific basis by the ISFSI operator. Section 10.2.2 indicates that the controlled area boundary dose as a result of an accident will not exceed 5 rem to any organ, exclusive of skin.

Table 1-2 NUREG-1536 Compliance Matrix (Continued)

Chapter 10 – Radiation Protection		
Area	**Regulatory Requirement**	**Description of Compliance**
4. ALARA	As a minimum, the proposed ALARA policy must fulfill the following criteria: a. To the extent practicable, the applicant should employ procedures and engineering controls that are founded upon sound radiation protection principles. b. Any design change should account for radiation protection, technological, and economical considerations. c. The applicant should have a written policy statement reflecting management commitment to maintain occupational and public exposures to radiation and radioactive material ALARA.	The description of the ALARA considerations of the system are provided in Section 10.1. The operating procedures provided in Chapter 8 are developed to keep occupational doses ALARA.

Table 1-2 NUREG-1536 Compliance Matrix (Continued)

Chapter 11 – Accident Analysis		
Area	**Regulatory Requirement**	**Description of Compliance**
1. Credible Accident and Natural Phenomena	Structures, systems, and components (SSC) important to safety must be designed to withstand credible accidents and natural phenomena without impairing their ability to perform safety functions. [10 CFR 72.24(d)(2); 10 CFR 72.122(b)(2), (3), (d), and (g)]	Analyses of the system for a variety of postulated off-normal and accident conditions are presented in Sections 11.1 and 11.2, respectively.
2. Controlled Area Boundary Dose	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 as a result of exposure to the sources listed in the regulations. [10 CFR 72.104(a); 10 CFR 72.236(d); and 10 CFR 72.24(d)]	The controlled area boundary dose calculations and minimum controlled area boundary distances under normal conditions are presented in Section 10.4.
3. Design Basis Accident Dose	Dose Limits for Design-Basis Accidents require that any individual located on or beyond the nearest boundary of the controlled area shall not receive a dose greater than 5 rem to the whole body or any organ from any design basis accident. [10 CFR 72.106(b); 10 CFR 72.24(m); and 10 CFR 72.24(d)(2)]	Section 10.2.2 indicates that the controlled area boundary dose as a result of an accident will not exceed 5 rem to any organ, exclusive of skin.
4. Criticality Control	The spent fuel must be maintained in a subcritical condition under credible conditions. [10 CFR 72.236(c) and 10 CFR 72.124(a)]	Section 6.4 presents the results of the criticality evaluation of the storage cask for the most reactive credible conditions, including the consequences of the off-normal and accident condition events evaluated in Sections 11.1 and 11.2, respectively.
5. Confinement Control	The cask and its systems important to safety must be evaluated, using appropriate tests or by other means acceptable to the Commission, to demonstrate that they will reasonably maintain confinement of radioactive material under credible accident conditions. [10 CFR 72.236(l)]	As stated in Section 7.3, the confinement system maintains its integrity for all credible off-normal and accident conditions.
6. Ready Retrieval of Spent Fuel	Storage systems must allow ready retrieval of spent fuel for further processing or disposal. [10 CFR 72.122(l)]	The off-normal and accident condition analyses presented in Sections 11.1 and 11.2 demonstrate that the spent fuel contents are protected during off-normal and accident conditions. Therefore, retrieval of the spent fuel from the system is not impacted by these postulated events.

Table 1-2 NUREG-1536 Compliance Matrix (Continued)

Chapter 11 – Accident Analysis		
Area	**Regulatory Requirement**	**Description of Compliance**
7. Monitoring Systems	Instrumentation and control systems must be provided to monitor systems that are important to safety over anticipated ranges for normal operation and off-normal operation. Those instruments and control systems that must remain operational under accident conditions must be identified in the Safety Analysis Report. [10 CFR 72.122(i)]	The system utilizes temperature monitoring instrumentation but utilizes routine inspection and surveillance to verify proper thermal operation of the system. The confinement system is fully welded and is leak tested to leaktight criteria as specified in Appendix 12A, Section 3.1.4. No seal monitoring is required.
8. Surveillance	Where instrumentation and control systems are not appropriate, storage confinement systems must have the capability for continuous monitoring in a manner such that the licensee will be able to determine when corrective action needs to be taken to maintain safe storage conditions. [72.122(h)(4)]	No active, continuous monitoring systems are required. Licensee radiological monitoring programs assure ISFSI operations meet 10 CFR 72.104 and 72.106 requirements.

Table 1-2 NUREG-1536 Compliance Matrix (Continued)

Chapter 11 – Accident Analysis		
Area	**Acceptance Criteria**	**Description of Compliance**
1. Dose Limits for Off-Normal Events	During normal operations and anticipated occurrences, the requirements specified in 10 CFR Part 20 must be met. In addition the annual dose equivalent to any individual located beyond the controlled area must not exceed 25 mrem to the whole body, 75 mrem to the thyroid, and 25 mrem to any other organ as a result of exposure to the following sources: a. planned discharges to the general environment of radioactive materials (with the exception of radon and its decay products) b. direct radiation from operations of the independent spent fuel storage installation (ISFSI) c. any other cumulative radiation from uranium fuel cycle operations (i.e., nuclear power plant) in the affected area	The controlled area boundary dose calculations and minimum controlled area boundary distances under normal conditions are presented in Section 10.4. No off-normal events are postulated that would result in a controlled area boundary dose in excess of the normal condition analysis.
2. Dose Limit for Design-Basis Accidents	Any individual located at or beyond the nearest controlled area boundary must not receive a dose greater than 5 rem to the whole body or any organ from any design-basis accident.	Section 10.2.2 indicates that the controlled area boundary dose as a result of an accident will not exceed 5 rem to any organ, exclusive of skin.
3. Criticality	The spent fuel must be maintained in a subcritical condition under credible conditions (i.e., k_{eff} equal to or less than 0.95). At least two unlikely, independent, and concurrent or sequential changes must be postulated to occur in the conditions essential to nuclear criticality safety before a nuclear criticality accident is possible (double contingency).	Section 6.4 presents the results of the criticality evaluation of the storage cask for the most reactive credible conditions, including the consequences of the off-normal and accident condition events evaluated in Sections 11.1 and 11.2, respectively. As stated in Section 6.1, the criticality analyses are performed for the most reactive credible configuration of the cask, at the highest enrichment, without credit for fuel burnup, and at the most reactive internal water moderator density, even though it is stated that water intrusion is not a credible event. Therefore, criticality cannot occur unless two separate events, such as (1) misloading a higher than design-basis enrichment, unirradiated fuel assembly and (2) water intrusion, occur.

Table 1-2 NUREG-1536 Compliance Matrix (Continued)

Chapter 11 – Accident Analysis		
Area	**Acceptance Criteria**	**Description of Compliance**
4. Confinement	The cask and its systems important to safety must be evaluated, using appropriate tests or by other means acceptable to the Commission, to demonstrate that they will reasonably maintain confinement of radioactive material under credible accident conditions.	As stated in Section 7.3, the confinement system maintains its integrity for all credible off-normal and accident conditions.
5. Retrievability	Retrievability is the capability to return the stored radioactive material to a safe condition without endangering public health and safety. This generally means ensuring that any potential release of radioactive materials to the environment or radiation exposures is not in excess of the limits in 10 CFR 20 or 10 CFR 72.122(h)(5). ISFSI and MRS storage systems must be designed to allow ready retrieval of the stored spent fuel or high level waste (MRS only) for compliance with 10 CFR 72.122(l).	The off-normal and accident condition analyses presented in Sections 11.1 and 11.2 demonstrate that the spent fuel contents are protected during off-normal and accident conditions. Therefore, retrieval of the spent fuel from the system is not impacted by these postulated events.
6. Instrumentation	The SAR must identify all instruments and control systems that must remain operational under accident conditions.	The system does not utilize instrumentation and control systems, but utilizes routine inspection and surveillance to verify proper operation of the system.

Table 1-2 NUREG-1536 Compliance Matrix (Continued)

Chapter 12 – Operating Controls and Limits	
Regulatory Requirement	**Description of Compliance**
1. General Requirement for Technical Specifications	
The applicant shall propose technical specifications (complete with acceptable bases and adequate justification). These specifications must include the following five areas [10 CFR 72.44(c), 10 CFR 72.24(g), and 10 CFR 72.26]:	
a. functional/operating limits, monitoring instruments, and limiting controls	Functional and operating limits are specified in Appendix 12A, Section 2.0.
b. limiting conditions	Limiting conditions for operation are specified in Appendix 12A, Section 3.0.
c. surveillance requirements	Surveillance requirements are specified in Appendix 12A, Section 3.0.
d. design features	Design features are specified in Appendix 12A, Section 4.0.
e. administrative controls	Administrative controls are specified in Section 12.5.
Subpart E, "Siting Evaluation Factors," and Subpart F, "General Design Criteria," to 10 CFR Part 72, provide the bases for the cask system design and, hence, are applicable as bases for appropriate technical specifications.	
2. Specific Requirements for Technical Specifications — Storage Cask Approval	
As a condition of approval, the design, fabrication, testing, and maintenance of a spent fuel DCSS must comply with the requirements of 10 CFR 72.236. [10 CFR 72.234(a)] 10 CFR 72.236 Specific Requirements for Spent Fuel Storage Cask Approval	The operating controls, limits, and surveillance activities specified in Chapter 12 are intended to ensure that the system is maintained within its design basis through all normal, off-normal, and accident conditions.

Table 1-2 NUREG-1536 Compliance Matrix (Continued)

Chapter 12 – Operating Controls and Limits	
Regulatory Requirement	**Description of Compliance**
The applicant must provide specifications for the spent fuel to be stored in the DCSS. At a minimum, these specifications should include, but not be limited to the following details [10 CFR 72.236(a)]: a. type of spent fuel (i.e., BWR, PWR, or both) b. maximum allowable enrichment of the fuel prior to any irradiation c. burn-up (i.e., megawatt-days/MTU) d. minimum acceptable cooling time of the spent fuel prior to storage in the DCSS (minimum 1 year) e. maximum heat that the DCSS system is designed to dissipate f. maximum spent fuel loading limit weights and dimensions h. condition of the spent fuel (i.e., intact assembly or consolidated fuel rods) i. inerting atmosphere requirements	Specifications for the spent fuel contents are provided in Appendix 12A, Tables 12A2-1 and 12A2-2. As specified in Appendix 12A, Section 3.1.3, the canister is backfilled with helium gas to maintain an inert atmosphere for the spent fuel.
The applicant must provide design bases and design criteria for structures, systems, and components (SSCs) important to safety. [10 CFR 72.236(b)]	The design bases and criteria for the system are specified in Section 2.2.
The applicant must design and fabricate the DCSS so that the spent fuel will be maintained in a subcritical condition under credible conditions. [10 CFR 72.236(c)]	As shown in Section 6.4, the spent fuel is maintained in a subcritical configuration under all credible configurations.
The applicant must provide radiation shielding and confinement features that are sufficient to meet the requirements in 10 CFR 72.104 and 72.106 regarding radioactive material in effluents, direct radiation, and area control. [10 CFR 72.236(d) and 10 CFR Part 20] 10 CFR 72.104 Criteria for Radioactive Materials in Effluents and Direct Radiation from an ISFSI or MRS 10 CFR 72.106 Controlled Area of an ISFSI or MRS	The maximum external dose rates for the system are specified in Appendix 12A, Section 3.2.1. These limits are established to ensure that, for the minimum controlled area boundary distance presented in Section 10.4, the controlled area boundary annual dose will be maintained within allowable limits.

Table 1-2 NUREG-1536 Compliance Matrix (Continued)

Chapter 12 – Operating Controls and Limits	
Regulatory Requirement	**Description of Compliance**
The applicant must design the DCSS to meet the following criteria:	
• Provide redundant sealing of confinement systems. [10 CFR 72.236(e)]	The redundant sealing features of the confinement system are presented in Section 2.3.2.1 and Chapter 7.
• Provide adequate heat removal capacity without active cooling systems. [10 CFR 72.236(f)]	As shown in Table 4.1-4, the system provides adequate heat removal through the passive cooling design features described in Section 4.1.
• Safely store the spent fuel for a minimum of 20 years and permit maintenance as required. [10 CFR 72.236(g)]	Section 1.1 and Table 2-1 specify a 50-year design life for the system. Routine maintenance is permitted as specified by Section 9.2.
• Facilitate decontamination to the extent practicable. [10 CFR 72.236(i)]	Decommissioning of the system is discussed in Section 2.4.
The DCSS must be compatible with wet or dry spent fuel loading and unloading facilities. [10 CFR 72. 236(h)]	The operating procedures for the system are presented in Chapter 8, and include procedures for wet and dry loading and unloading operations.
The applicant must inspect the DCSS to ascertain that there are no cracks, pinholes, uncontrolled voids, or other defects that could significantly reduce its confinement effectiveness. [10 CFR 72.236(j)]	As described in Section 9.1.1, the canister is visually and non-destructively examined prior to use.
The applicant must evaluate the DCSS, 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 72.236(l)]	The canister is analyzed for normal conditions in Section 3.4.4.1, and for off-normal and accident conditions in Sections 11.1 and 11.2, respectively. Because the canister maintains adequate positive margins of safety, the system will reasonably maintain confinement under all credible conditions.

Table 1-2 NUREG-1536 Compliance Matrix (Continued)

Chapter 13 – Quality Assurance	
Regulatory Requirement	**Description of Compliance**
According to 10 CFR 72.24, "Contents of Application: Technical Information," the application must include, at a minimum, a description that satisfies the requirements of 10 CFR Part 72, Subpart G, "Quality Assurance," with regard to the QA program to be applied to the design, fabrication, construction, testing, and operation of the DCSS SSCs important to safety. Moreover, Subpart G states that the licensee shall establish the QA program at the earliest practicable time consistent with the schedule for accomplishing the activities.	A synopsis of the NAC Quality Assurance Program is presented in Section 13.2. This program description is consistent with the 18 criteria specified in Subpart G. The Quality Assurance Program is approved by the NRC under 10 CFR 71, Subpart H.

Table 1-2 NUREG-1536 Compliance Matrix (Continued)

Chapter 13 – Quality Assurance		
Area	**Acceptance Criteria**	**Description of Compliance**
1. Quality Assurance Organization	The SAR should describe (and illustrate in an appropriate chart) the organizational structure, interrelationships, and areas of functional responsibility and authority for all organizations performing quality- and safety-related activities, including both the applicant's organization and principal contractors, if applicable. Persons or organizations responsible for ensuring that an appropriate QA program has been established and verifying that activities affecting quality have been correctly performed should have sufficient authority, access to work areas, and organizational freedom to carry out that responsibility.	The QA organization is described in Section 13.2.1. An organizational chart is provided in Figure 13.2-1.
2. Quality Assurance Program	The SAR should provide acceptable evidence that the applicant's proposed QA program will be well-documented, planned, implemented, and maintained to provide the appropriate level of control over activities and SSCs, consistent with their relative importance to safety.	The implementation of the QA program is described in Section 13.2.2.
3. Design Control	The SAR should describe the approach that the applicant will use to define, control, and verify the design and development of the DCSS. An effective design control program will provide assurance that the proposed DCSS will be appropriately designed and tested and will perform its intended function.	Design control is described in Section 13.2.3.
4. Procurement Document Control	Documents used to procure SSCs or services should include or reference applicable design bases and other requirements necessary to ensure adequate quality. To the extent necessary, these procurement documents should require that suppliers have a QA program consistent with the quality level of the SSCs or services to be procured.	Procurement document control is described in Section 13.2.4.
5. Instructions, Procedures, and Drawings	The SAR should define the applicant's proposed procedures for ensuring that activities affecting quality will be prescribed by, and performed in accordance with, documented instructions, procedures, or drawings of a type appropriate for the circumstances.	Procedures, instructions and drawings are described in Section 13.2.5.
6. Document Control	The SAR should define the applicant's proposed procedures for preparing, issuing, and revising documents that specify quality requirements or prescribe activities affecting quality. These procedures should provide adequate control to ensure that only the latest documents are used. In addition, the applicant's authorized personnel should carefully review and approve the accuracy of all documents and associated revisions before they are released for use.	Document control is described in Section 13.2.6.

Table 1-2 NUREG-1536 Compliance Matrix (Continued)

Chapter 13 – Quality Assurance		
Area	**Acceptance Criteria**	**Description of Compliance**
7. Control of Purchased Material, Equipment, and Services	The SAR should define the applicant's proposed procedures for controlling purchased material, equipment, and services to ensure conformance with specified requirements.	Control of purchased items and services is described in Section 13.2.7.
8. Identification and Control of Materials, Parts, and Components	The SAR should define the applicant's proposed provisions for identifying and controlling materials, parts, and components to ensure that incorrect or defective SSCs are not used.	Identification and control of material, parts and components are described in Section 13.2.8.
9. Control of Special Processes	The SAR should describe the controls that the applicant will establish to ensure the acceptability of special processes (such as welding, heat treatment, nondestructive testing, and chemical cleaning) and that they are performed by qualified personnel using qualified procedures and equipment.	Control of special processes is described in Section 13.2.9.
10. Licensee Inspection	The SAR should define the applicant's proposed provisions for inspection of activities affecting quality to verify conformance with instructions, procedures, and drawings.	Inspection is described in Section 13.2.10.
11. Test Control	The SAR should define the applicant's proposed provisions for tests to verify that SSCs conform to specified requirements and will perform satisfactorily in service. The applicant should specify test requirements in written procedures, including provisions for documenting and evaluating test results. In addition, the applicant should establish qualification programs for test personnel.	Test control is described in Section 13.2.11.
12. Control of Measuring and Test Equipment	The SAR should define the applicant's proposed provisions to ensure that tools, gauges, instruments, and other measuring and testing devices are properly identified, controlled, calibrated, and adjusted at specified intervals.	Control of measuring and test equipment is described in Section 13.2.12.
13. Handling, Storage, and Shipping Control	The SAR should define the applicant's proposed provisions to control the handling, storage, shipping, cleaning, and preservation of SSCs in accordance with work and inspection instructions to prevent damage, loss, and deterioration.	Handling, storage and shipping are described in Section 13.2.13.

Table 1-2 NUREG-1536 Compliance Matrix (Continued)

Chapter 13 – Quality Assurance		
Area	**Acceptance Criteria**	**Description of Compliance**
14. Inspection, Test, and Operating Status	The SAR should define the applicant's proposed provisions to control the inspection, test, and operating status of SSCs to prevent inadvertent use or bypassing of inspections and tests.	Inspection, test, and operating status are described in Section 13.2.14.
15. Nonconforming Materials, Parts, or Components	The SAR should define the applicant's proposed provisions to control the use or disposition of nonconforming materials, parts, or components.	Control of nonconforming items is described in Section 13.2.15.
16. Corrective Action	The SAR should define the applicant's proposed provisions to ensure that conditions adverse to quality are promptly identified and corrected and that measures are taken to preclude recurrence.	Corrective action is described in Section 13.2.16.
17. Quality Assurance Records	The SAR should define the applicant's proposed provisions for identifying, retaining, retrieving, and maintaining records that document evidence of the control of quality for activities and SSCs important to safety.	Records are described in Section 13.2.17.
18. Audits	The SAR should define the applicant's proposed provisions for planning, scheduling, and conducting audits to verify compliance with all aspects of the QA program, and to determine the effectiveness of the overall program. The SAR should clearly identify responsibilities and procedures for conducting audits, documenting and reviewing audit results, and designating management levels to review and assess audit results. In addition, the SAR should describe the applicant's provisions for incorporating the status of audit recommendations in management reports.	Audits are described in Section 13.2.18.

Table 1-2 NUREG-1536 Compliance Matrix (Continued)

Chapter 14 – Decommissioning		
Area	**Regulatory Requirement**	**Description of Compliance**
1. Facility Design Features	The ISFSI or MRS must be designed for decommissioning. Provisions must be made to facilitate decontamination of structures and equipment, minimize the quantity of radioactive wastes and contaminated equipment, and facilitate the removal of radioactive wastes and contaminated materials at the time the ISFSI or MRS is permanently decommissioned. [10 CFR 72.130.]	The design of the ISFSI facility is site-specific, and thus not applicable to a DCSS. Decommissioning considerations are discussed in Section 2.4.
2. Cask Design Features	The cask must be designed to facilitate decontamination to the extent practicable. [10 CFR 72.236(i).]	The decontamination features of the system are discussed in Section 2.4.
3. Financial / Records	The requirements for financial assurance and record keeping associated with decommissioning are found in 10 CFR 72.30. 10 CFR 72.30 Financial Assurance and Recordkeeping for Decommissioning	Financial and record keeping issues are site-specific, and thus not applicable to a DCSS.
4. License Termination	The requirements for terminating an ISFSI license and decommissioning ISFSI sites and buildings are found in 10 CFR 72.54, including the requirements for submitting the final decommissioning plan.	ISFSI license termination is a site-specific issue, and thus not applicable to a DCSS.

Chapter 14 – Decommissioning	
Acceptance Criteria	**Description of Compliance**
1. Decontamination of buildings and equipment, as specified in RG 1.86.	The decontamination features of the system are discussed in Section 2.4.
2. Classification and disposal of wastes, as contained in 10 CFR 61.55.	Not applicable.

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1.1 Introduction

The NAC-MPC system is a transport compatible dry storage system that uses a vertical concrete storage cask and a stainless steel transportable storage canister (canister) with a welded closure to safely store irradiated nuclear fuel (spent fuel). The canister is stored in the central cavity of the concrete cask and is compatible with the NAC-STC transport cask for future off-site shipment. The concrete storage cask provides radiation shielding and contains internal air flow paths that allow the decay heat from the canister contents to be removed by natural air circulation around the canister wall. The system is designed to meet the requirements for storage of Yankee Class spent fuel. The NAC-MPC is designed and analyzed for a 50-year life.

The principal components of the NAC-MPC system are the canister, the vertical concrete cask and the transfer cask. The loaded canister is moved to and from the concrete cask with the transfer cask. The transfer cask provides radiation shielding while the canister is being closed and sealed and while the canister is being transferred. The canister is placed in the concrete cask by positioning the transfer cask with the loaded canister on top of the concrete cask and lowering the canister into the concrete cask. Figure 1.1-1 depicts the major components of the NAC-MPC system and shows the transfer cask positioned on the top of the concrete cask.

The NAC-MPC is designed to safely store up to 36 intact Yankee Class spent fuel and reconfigured fuel assemblies. The fuel is initially loaded into a canister containing a fuel basket. Figure 1.1-2 depicts the canister and the spent fuel basket. The design characteristics of the NAC-MPC are shown in Table 1.1-1.

Yankee Class fuel includes United Nuclear, Combustion Engineering, Exxon-ANF, and Westinghouse Type A and Type B fuel designs. The Type A and Type B fuel designs are complementary configurations that accommodate the use of a cruciform control blade in reactor operations. The fuel specifications that serve as the design basis for the NAC-MPC are presented in Section 2.1.

The system design and analyses were performed in accordance with Title 10, Code of Federal Regulations, Part 72 (10 CFR 72), ANSI/ANS 57.9-1984 and the applicable sections of the ASME Boiler and Pressure Vessel Code and the American Concrete Institute Code.

Figure 1.1-1 Major Components of the NAC-MPC System

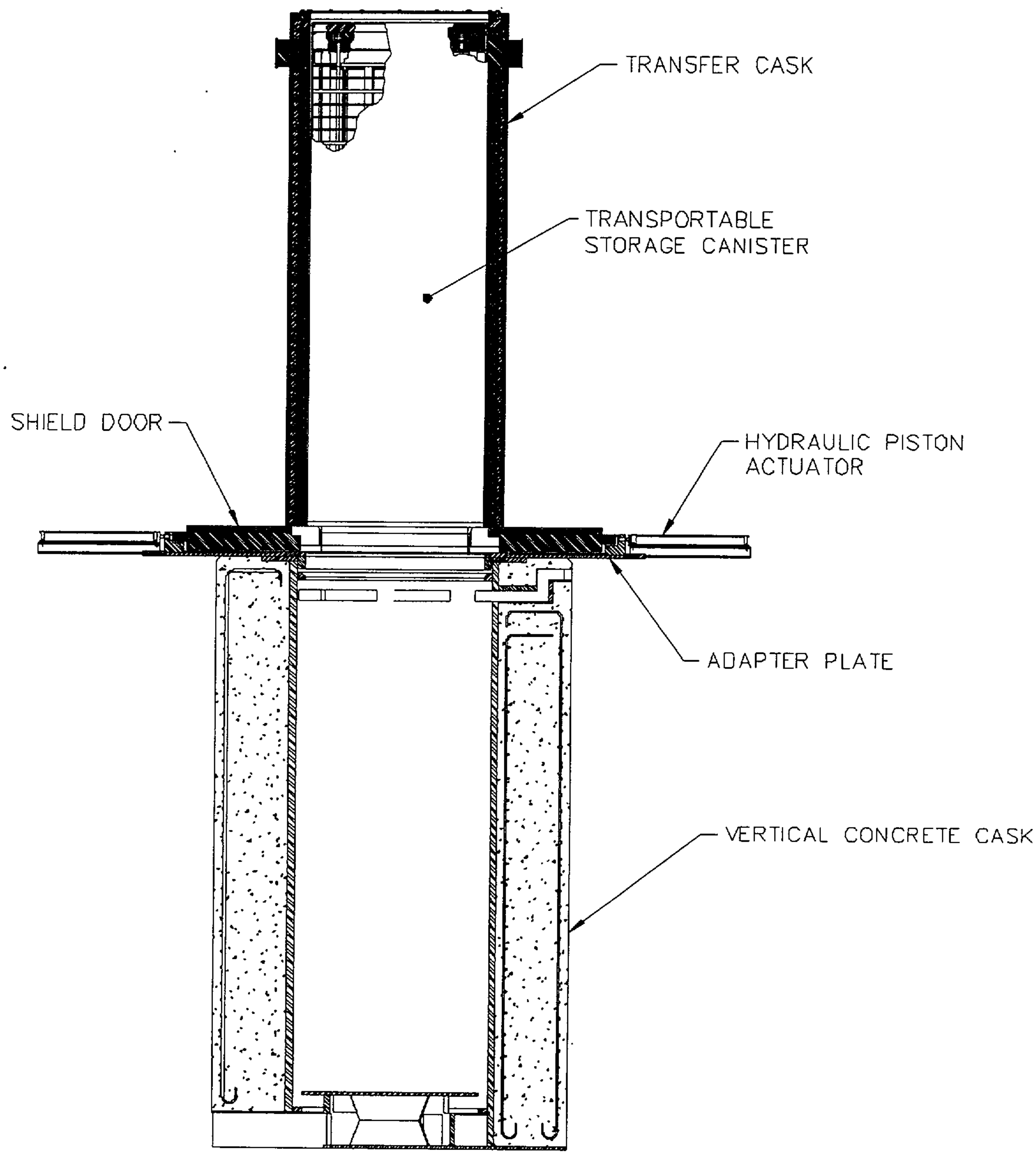

Figure 1.1-2 Transportable Storage Canister Showing the Spent Fuel Basket

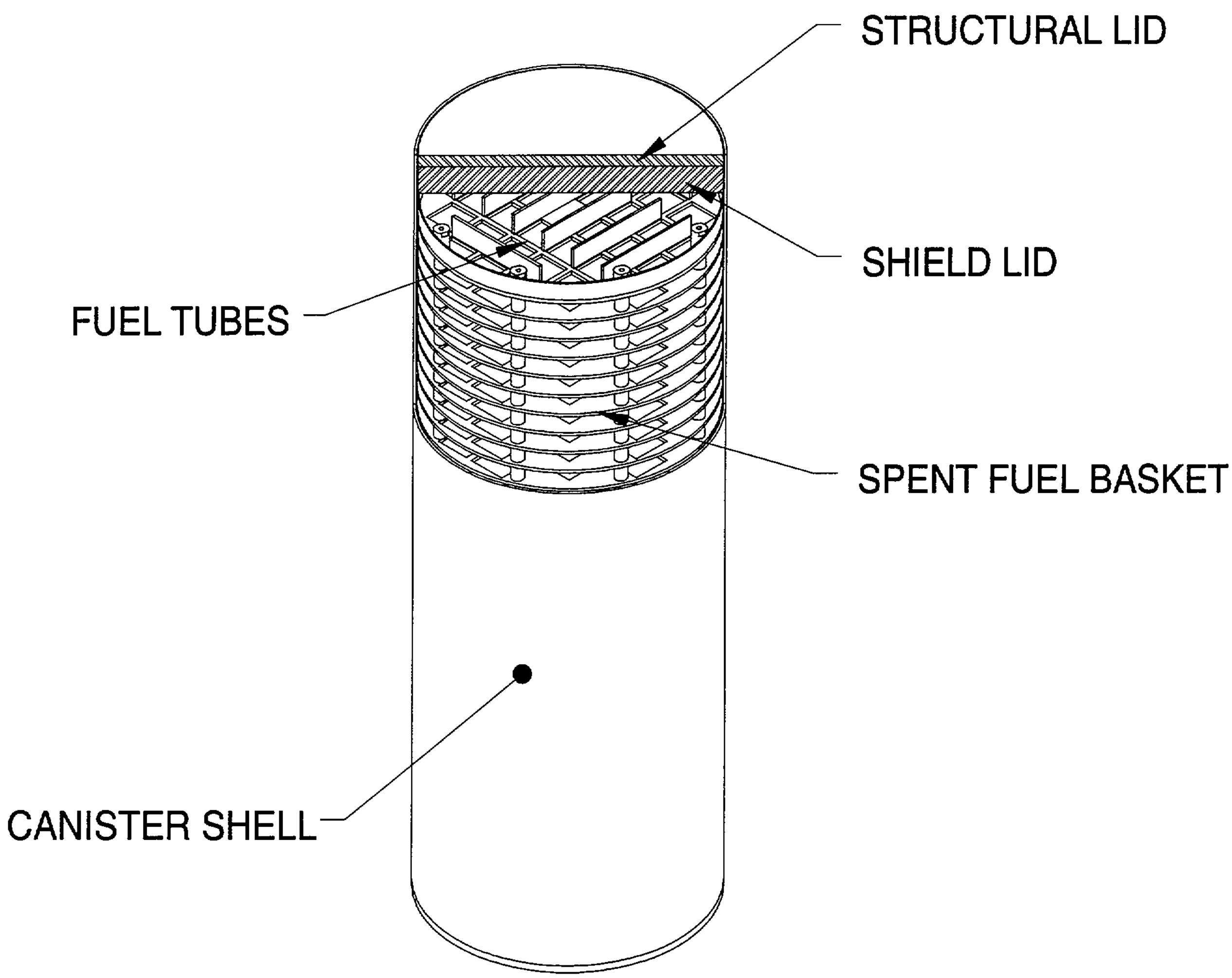

Table 1.1-1 Design Characteristics of the NAC-MPC

Design Characteristic	Dimension[1]	Material
Canister		
- Shell	5/8 thick Plate	Type 304L Stainless Steel
- Bottom	1.0 thick Plate	Type 304L Stainless Steel
- Shield Lid	5.0 thick Plate	Type 304 Stainless Steel
- Structural Lid	3.0 thick Plate	Type 304L Stainless Steel
Canister Fuel Basket		
- End Weldments	0.5 x 68.98 dia.	Type 304 Stainless Steel
- Support Disks	0.5 x 69.15 dia.	Type 17-4 PH Stainless Steel
- Heat Transfer Disks	0.5 x 68.70 dia.	Type 6061-T6 Aluminum Alloy
- Tube	7.80 x 7.80 x 0.048	Type 304 Stainless Steel encasing BORAL
- Spacers	2.5 diameter	Type 304 Stainless Steel
- Tie Rods (8)	1-1/8 diameter	Type 304 Stainless Steel

1. Dimensions in inches unless otherwise noted.

Table 1.1-1 Design Characteristics of the NAC-MPC (Continued)

Design Characteristic	Dimension[1]	Material
Transfer Cask		
- Outer Shell	1.5 x 86.5 dia.	ASTM A588 Low Alloy Steel
- Inner Shell	0.75 x 73.0 dia.	ASTM A588 Low Alloy Steel
- Retaining Ring	0.75 x 80.8 dia.	ASTM A588 Low Alloy Steel
- Trunnions	10.0 dia.	A350 LF2 Carbon Steel
- Bottom Plate	1.0 thick Plate	ASTM A588 Low Alloy Steel
- Top Plate	2.0 thick Plate	ASTM A588 Low Alloy Steel
-Shield Doors	9 1/2 thick Plate	A350 LF2 Carbon Steel
- Door Rails	9.88 x 6.5	A350 LF2 Carbon Steel
- Gamma Shield	3.5 thick	ASTM B29, Chem. Grade Lead
- Neutron Shield	2.0 thick	NS-4-FR, Solid Synthetic Polymer
Transfer Adapter		
- Base Plate	2 thick Plate	ASTM A36 Carbon Steel
- Locating Ring	2.0 wide x 78.25 dia.	ASTM A36 Carbon Steel

1. Dimensions in inches unless otherwise noted.

Table 1.1-1 Design Characteristics of the NAC-MPC (Continued)

Design Characteristic	Dimension[1]	Material
Vertical Concrete Cask		
Weldment Structure		
- Shell	3.5 thick x 86.0 dia.	ASTM A36 Carbon Steel
- Top Flange	2.0 thick x 97.9 dia.	ASTM A36 Carbon Steel
- Support Ring	2.5 thick x 79.0 dia.	ASTM A36 Carbon Steel
- Base Plate	2.0 thick x 72.0 dia.	ASTM A36 Carbon Steel
Concrete Cask		
- Concrete Shell	21.0 thick x 128 dia.	Type II Portland Cement
- Shield Plug	5.13 x 78.5 dia.	ASTM A36, Carbon Steel NS-4-FR
- Top Plate	1.5 thick x 92.1 dia.	ASTM A36, Carbon Steel
- Rebar	Various	ASTM A615, GR 60, Carbon Steel

1. Dimensions in inches unless otherwise noted.

1.2 The NAC-MPC System

The NAC-MPC system provides long-term storage and subsequent transport of Yankee Class spent fuel using the certified NAC-STC. During long-term storage, the system provides an inert environment; passive shielding, cooling, and criticality control; and, a confinement boundary closed by welding. The structural integrity of the system precludes the release of contents in any of the design basis normal conditions and off-normal or accident events, thereby assuring public health and safety during use of the system.

1.2.1 NAC-MPC System Components

The NAC-MPC system consists of three principal components:

- Transportable storage canister (canister),
- Vertical concrete cask, and
- Transfer cask.

Ancillary equipment needed to use the NAC-MPC System is:

- Automated or manual welding equipment;
- An air pallet or hydraulic roller skid (used to move the storage cask on and off the heavy haul transfer trailer and to position the storage cask on the storage pad);
- Suction pump, vacuum drying, helium backfill and leak detection equipment;
- A heavy haul trailer or cask transporter (for storage cask transport to the storage pad);
- Adapter plate and hardware to position the transfer cask with respect to the storage or transport cask; and,
- A lifting yoke for the transfer cask and lifting slings for the canister and canister lids.

In addition to these items, the system requires utility services (electric, air and water), common tools and fittings, and miscellaneous hardware.

The transportable storage canister is designed to be transported in the NAC-STC Storable Transport Cask (Certificate of Compliance No. 71-9235). The transport load conditions produce higher stresses in the canister than would be produced by the storage load conditions alone.

Consequently, the canister design is conservative with respect to storage conditions. The evaluation of the canister for transport conditions is found in the Safety Analysis Report for the NAC-STC Storable Transport Cask, Docket No. 71-9235.

1.2.1.1 Transportable Storage Canister and Baskets

The transportable storage canister contains a basket that accommodates up to 36 intact Yankee Class spent fuel assemblies and reconfigured fuel assemblies (RFA) up to a total contents weight of 30,600 pounds.

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 boundaries. The fuel basket is based on the directly loaded fuel basket design used in the certified NAC-STC. This basket features the NAC-patented poison tubes and stacked disk design with heat transfer disks. The basket was analyzed using the ANSYS computer code to demonstrate that it can withstand the horizontal drop loads without deforming in a way that damages or constrains a fuel assembly. This tube and disk design has been accepted and approved by the NRC, pursuant to 10 CFR 71 and 10 CFR 72. Table 1.2-1 summarizes the major physical design parameters of the canister.

The fuel basket design is a right-circular cylinder configuration with 36 fuel tubes laterally supported by a series of support disks, which are retained by spacers on eight radially located tie rods. The support disks are stainless steel (17-4 PH) with holes for the poison fuel tubes. The basket top and bottom weldments are fabricated from Type 304 stainless steel. The tie rods and spacer sleeves are also fabricated from Type 304 stainless steel. The fuel assemblies are contained in fuel tubes. The fuel tubes are fabricated from Type 304 stainless steel with encased BORAL sheets on all four outside surfaces of the fuel tube. The BORAL provides criticality control in the basket.

The heat transfer disks are aluminum with holes for the fuel tubes. The heat transfer disks are spaced midway between the support disks and are the primary path for conducting the heat from the fuel assemblies to the canister wall. Holes in the heat transfer disks for the tubes and tie rods are sized to accommodate thermal expansion occurring after the fuel is placed into the basket.

The fuel basket tube and disk design provides the structural integrity to maintain the spent fuel in a subcritical configuration during normal operations and the hypothetical accident events, even if optimum moderator condition and fresh fuel are assumed. With the most reactive fuel, the fuel basket maintains $K_{eff} \leq 0.95$. Subcriticality is assured assuming fresh fuel loading and no soluble boron in the spent fuel pool water during fuel loading operations.

The transportable storage canister assembly is designed to facilitate filling with water and subsequent draining and drying. A rounded notch is located at the bottom of each fuel tube, ensuring free flow of water between the inner tube regions and the disk regions. Each of the disks also has three holes to supplement the flow of water between disks. In addition, the bottom plate is positioned by supports above the bottom of the canister to facilitate water flow to the drain line.

The canister is fabricated from 5/8-inch thick Type 304L stainless steel rolled 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 stainless steel material was selected to be compatible with the DOE MPC program guidelines for future disposal and to minimize the potential for any adverse chemical reactions in the spent fuel pool. The design of the shield lid and structural lid allows a redundant confinement seal at the top of the canister. Each lid weld is liquid penetrant examined.

The 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 dry storage. The drain port is equipped with a quick disconnect fitting and a drain tube that extends nearly to the bottom of the canister. The vent port extends to the underside of the shield lid and is equipped with a quick disconnect fitting used for vacuum drying and helium backfilling. After draining, drying, backfilling, and testing operations are complete, port covers are installed and welded to the shield lid to seal the penetration.

A summary of the canister fabrication specifications is presented in Table 1.2-2.

1.2.1.2 Vertical Concrete Cask

The vertical concrete cask (storage cask) is the storage overpack for the transportable storage canister. It provides structural support, shielding, protection from environmental conditions, and natural convection cooling of the canister during long-term storage. Table 1.2-3 lists the principal physical design parameters of the storage cask. The storage cask is a reinforced concrete (Type II Portland cement) structure with a structural 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 design basis fuel. Inner and outer reinforcing steel (rebar) assemblies are contained within the concrete. The reinforced concrete wall provides the structural strength to protect the canister and its contents in natural phenomena events such as tornado wind loading and wind driven missiles. The storage cask incorporates reinforced chamfered corners at the edges to facilitate construction. "Fire block," an insulating material provided by BISCO, is placed on the base of the cavity to prevent contact between the stainless steel canister and the carbon steel pedestal. The storage cask is shown in Figure 1.2-1.

The storage cask has an annular air passage to allow the natural circulation of air around the canister to remove the decay heat from the spent fuel. The air inlet and outlet vents are steel-lined penetrations that take nonplanar paths to the concrete cask cavity to minimize radiation streaming. The decay heat is transferred from the fuel assemblies to the tubes in the fuel basket and through the heat transfer disks to the canister wall. Heat flows by radiation and convection from the canister wall to the air circulating through the concrete cask annular air passage and is exhausted through the air outlet vents. This passive cooling system is designed to maintain the peak cladding temperature of both stainless steel and zircaloy clad fuel well below acceptable limits during long-term storage. This design also maintains the bulk concrete temperature below 150°F and localized concrete temperatures below 200°F in normal operating conditions.

The top of the storage cask is closed by a shield plug and lid. The shield plug is approximately 5 inches thick and incorporates carbon steel plate as gamma radiation shielding and NS-4-FR as neutron radiation shielding. A carbon steel lid that provides additional gamma radiation shielding is installed above the shield plug. The shield plug and lid reduce skyshine radiation and provide a cover and seal to protect the canister from the environment and postulated tornado missiles. The lid is bolted in place and has tamper indicating seals on two of the installation bolts.

Fabrication of the storage cask involves no unique or unusual forming, concrete placement, or reinforcement requirements. The concrete portion of the storage cask is constructed by placing concrete between a reusable, exterior form and the inner metal liner. Reinforcing bars are used at the inner and outer concrete surfaces to provide structural integrity. The inner liner and base of the storage cask are shop fabricated.

The principal fabrication specifications for the storage cask are shown in Table 1.2-4.

1.2.1.3 Transfer Cask

The transfer cask, with its lifting yoke, is primarily a lifting device used to move the canister assembly. It provides biological shielding when it contains a loaded canister. The transfer cask is used for the vertical transfer of the canister between work stations and the storage cask, or transport cask. The general arrangement of the transfer cask and canister is shown in Figure 1.2-5, and the arrangement of the transfer cask and concrete cask is shown in Figure 1.2-6. The configuration of the transfer cask, canister and concrete cask during loading of the concrete cask is shown in Figure 1.2-7.

Table 1.2-5 shows the principal design parameters of the transfer cask.

The transfer cask is a multiwall (steel/lead/NS-4-FR neutron shield/steel) design, which limits the average contact radiation dose rate to less than 200 mrem/hr. The transfer cask design incorporates a top retaining ring, which is bolted in place, that prevents a loaded canister from being inadvertently removed through the top of the transfer cask. The transfer cask has retractable bottom shield doors. During loading operations, the doors are closed and secured by pins, so they cannot inadvertently open. During unloading, the doors are retracted using hydraulic cylinders to allow the canister to be lowered into the storage or transport casks. The transfer cask is shown in Figure 1.2-2.

To qualify the transfer cask as a heavy lifting device, it is designed, fabricated, and load tested to the requirements of NUREG-0612 and ANSI N14.6.

To minimize potential contamination during loading operations in the spent fuel pool, clean water is circulated in the gap between the transfer cask interior surface and the canister exterior surface. The transfer cask has two supply and two discharge lines passing through its wall.

These lines allow hoses to be connected and clean water to be pumped into and through the annular gap to preclude the intrusion of pool water when the canister is being loaded.

Exposed surfaces of the transfer cask, other than the load-bearing surfaces of the trunnions and the bottom door rails, are coated with Keeler & Long E-Series epoxy enamel to protect the carbon steel and to provide a smooth surface to facilitate decontamination.

1.2.1.4 Auxiliary Equipment

This section presents a brief description of the principal auxiliary equipment needed to operate the NAC-MPC in accordance with its design.

1.2.1.4.1 Adapter Plate

The adapter plate is a carbon steel table that bolts to the top of the vertical concrete (storage) cask or the NAC-STC and mates the transfer cask to either of those casks. It has a large center hole that allows the transportable storage canister to be raised or lowered through the plate into or out of the transfer cask. Rails are incorporated in the adapter plate to guide and support the bottom shield doors of the transfer cask when they are in the open position. The adapter plate also supports the hydraulic system and the actuators that open and close the transfer cask bottom doors.

1.2.1.4.2 Air Pad Rig Set

The air pad rig set (air pad set) is a commercially available device, sometimes referred to as an air pallet. When inflated, the air pad set lifts the concrete cask by using high volume air. The air pads employ a continuous, regulated air flow and a control system that equalizes lifting heights of the four air pads by regulating compressed air flow to each of the air pads. The compressed air supply creates an air filler between the inflated air cushion and the supporting surface. The thin film of air allows the concrete cask to be lifted and moved. Once lifted, a suitable towing vehicle, such as a commercial tug or forklift can move the storage cask.

1.2.1.4.3 Automatic Welding System

The automatic welding system consists of commercially available components with a customized weld head. The components include: a welding machine, a remote pendant, a carriage, a drive motor and welding wire motor, and the weld head. The system is designed to make at least one weld pass automatically around the canister after its weld tip is manually positioned at the proper location. As a result, radiation exposure during canister closure is much less than would be incurred from manual welding.

1.2.1.4.4 Draining and Drying System

The draining and drying system consists of a suction pump and a vacuum pump. The suction pump is used to remove free water from the canister cavity. The vacuum pump is a two-stage unit for drying the interior of the canister. The first stage is a large capacity or "roughing" pump intended to remove free water not removed by the suction pump. The second stage is a vacuum pump used to evacuate the canister interior of the small amounts of remaining moisture and establish the vacuum condition.

1.2.1.4.5 Helium Leak Test Equipment

A helium leak detector is required to verify the integrity of the welds of the canister shield lid. The helium leak detector is the mass spectrometer type.

1.2.1.4.6 Heavy-Haul Trailer

The heavy haul trailer is used to move the vertical concrete storage cask. A special trailer has been designed for transport of the empty or loaded storage cask. The design incorporates a built-in jacking system that facilitates raising the storage cask to allow installation of the air pad set used to move the cask onto the storage pad. The trailer incorporates both reinforcing to increase the trailer load-bearing area and design features that reduce its turning radius. However, any commercial double-drop-frame trailer having a deck height approximately matching that of the storage pad could be used.

1.2.1.4.7 Lifting Jacks

Hydraulic jacks are installed at jacking pads in the bottom air ducts to lift the storage cask so that the air pad set can be installed or removed. Four hydraulic pad jacks are provided, along with a control panel, an electric hydraulic oil pump, an oil reservoir tank and all hydraulic lines and fittings. The jacks are used to lift the cask approximately three inches. This permits installation of four air pads under the concrete cask.

1.2.1.4.8 Rigging and Slings

Load rated rigging attachments and slings are provided for major components. The rigging attachments are swivel hoist rings that allow attachment of the slings to the hook. All slings are commercially purchased to have adequate safety margin to meet the requirements of ANSI N14.6 and NUREG 0612. The slings include a concrete cask lid sling, concrete cask shield plug sling, and canister shield lid sling, loaded canister transfer sling (also used to handle the structural lid), canister retaining ring sling. The appropriate rings or eye bolts are provided to accommodate each sling and component.

The transfer cask lifting yoke is specially designed and fabricated for lifting the transfer cask. It is designed to meet the requirements of ANSI N14.6 and NUREG 0612. It is single-failure-proof by design. The transfer cask lifting yoke is initially load tested to 300 percent of the design load.

1.2.1.4.9 Temperature Instrumentation

The vertical concrete cask has four outlet vents near the top of the cask and four inlet vents at the bottom. Each outlet is equipped with a permanent remote temperature detector mounted in the outlet air plenum. The detector is used to measure the outlet air temperature, which can be read at a display device located on the outside surface of the concrete cask or at a remote location. The detectors are installed on all of the storage casks at the ISFSI. One inlet of one concrete cask, or the ambient temperature of the ISFSI site, is also monitored.

1.2.1.5 Transport Cask

The transportable storage canister is designed to be transported in the NAC-STC. The canister is positioned in the NAC-STC cavity with two axial spacers. The spacers are required because the transport cask cavity length is 165 inches, while the length of the canister is 122.5 inches.

The NAC-STC is licensed by the NRC pursuant to 10 CFR 71 (Certificate of Compliance No. 71-9235). A request for an amendment to the NAC-STC Certificate of Compliance to incorporate transport of the canister was submitted to the NRC on December 30, 1996. The NAC-STC is designed for free interchange/rail shipment.

The transport configuration of the NAC-STC is shown in Figure 1.2-3.

1.2.2 Operational Features

This section outlines the principal handling activities of the NAC-MPC storage system. The system provides passive long-term storage of spent fuel in an inert environment. In storage, the only active system is for temperature monitoring of the outlet air. This temperature is recorded daily as a check of the thermal performance of the storage system design. This system does not penetrate the confinement boundary and is not essential to the safe operation of the NAC-MPC.

The principal activities associated with the use of the system are closing the canister and loading the canister in the storage cask. The transfer cask is designed to meet the requirements of these operations. The transfer cask holds the canister during loading with fuel; provides biological shielding during closing of the canister; and provides the means by which the loaded canister is moved to, and installed in, the storage cask. The canister assembly consists of five principal components: the canister shell (side wall and bottom), the shield lid, the vent port, the drain port (together with the vent and drain port covers), and the structural lid. A drain tube extends from the shield lid drain port to the bottom of the canister. The location of the drain and vent ports is shown in Figure 8.1-1.

The vent and drain ports allow the draining, vacuum drying, and backfilling with helium necessary to provide a dry, inert atmosphere for the contents. The vent and drain port covers, the shield lid, the canister shell, and the joining welds form the primary confinement boundary. This boundary is shown in Figure 7.1-1. A secondary confinement boundary is formed over the shield lid by the structural lid and the weld that joins it to the canister shell. This boundary is shown in Figure 7.1-2.

The structural lid contains the drilled and tapped holes for attachment of the swivel hoist rings used to lift the loaded canister. The drilled and tapped holes are filled with bolts or plugs to avoid collecting debris, and to preclude the possibility of radiation streaming from the holes, when the hoist rings are not installed.

The step-by-step procedures for use of the NAC-MPC system are presented in Chapter 8. The following list presents a brief description of the principal activities. This list assumes that the empty canister is installed in the transfer cask for spent fuel pool loading.

- Lift the transfer cask over the pool and start the flow of water to the transfer cask annulus and canister. After the annulus and canister are filled, lower the cask to the bottom of the pool.
- Load the selected spent fuel assemblies into the canister and set the shield lid.
- Raise the transfer cask from the pool. Decontaminate the transfer cask exterior as it clears the pool surface. Drain the annulus. Place the transfer cask in the decontamination area.
- Weld the shield lid to the canister shell. Pressure test the weld. Drain the pool water from the canister. Attach the vacuum system to the drain line, and operate the system to achieve a vacuum.
- Hold the vacuum and backfill with helium to 1 atmosphere. Restart the vacuum system and remove the helium. After achieving vacuum, backfill and pressurize the canister with helium.
- Helium leak check the shield lid weld. Vent the helium pressure to 1 atmosphere (absolute). Install the vent and drain port covers and weld them to the shield lid.
- Install the structural lid and weld it to the canister shell. Install the hoist rings, and attach the canister lifting sling. Install the adapter plate on the storage cask.
- Lift the transfer cask to the top of the storage cask and set it on the adapter plate, ensuring that the bottom door hydraulic actuators are engaged.
- Attach the canister lifting slings to the crane hook and lift the canister.
- Open the bottom doors of the transfer cask.
- Lower the canister into the storage cask. Detach the canister slings from the hook.
- Remove the transfer cask and adapter plate. Remove the canister lifting slings.
- Install the shield plug and lid on the concrete cask.
- Move the loaded storage cask to the storage pad.
- Using the air pad rig set and a towing vehicle, move the storage cask to its designated location on the storage pad.

The removal operations are essentially the reverse of these steps, except that weld removal and cool down of the contents are required.

The special equipment needed to operate the NAC-MPC system has been described in Section 1.2.1.4. Other items required are miscellaneous hardware, connection hose and fittings, and hand tools typically found at a reactor site.

1.2.3 Cask Contents

The NAC-MPC is designed to store up to 36 Yankee Class spent fuel assemblies. The Yankee Class fuel consists of fuel assemblies manufactured by Westinghouse, United Nuclear, Exxon, and Combustion Engineering. The assemblies vary in initial enrichment from 3.5 to 4.94 wt % ^{235}U. Each manufacturer's types of assemblies include two configurations identified as Type A and Type B. The arrangement of fuel rods differ in each types to allow the fuel assembly to accept a segment of a control blade used for criticality control. The characteristics of the Yankee Class spent fuels are presented in Table 1.2-6. Unenriched fuel assemblies are not evaluated and are not included as a proposed contents.

A canister may contain one or more Reconfigured Fuel Assemblies (RFA) designed to confine Yankee Class intact or damaged spent fuel rods and fuel debris. The RFA is designed to confine its contents during all storage and transport conditions. The assembly can accept up to 64 full length spent fuel rods in an eight by eight array of tubes. A sketch of the assembly is provided in Figure 1.2-4.

The Reconfigured Fuel Assembly consists of a shell (square tube with end fittings), a basket assembly and 64 fuel tubes. The external dimensions of the shell are the same as those of a standard Yankee Class spent fuel assembly and all materials are stainless steel. It is designed such that it can be handled in the same manner as a standard Yankee Class spent fuel assembly. The spent fuel is confined in the fuel tubes. The tubes are supported by a basket assembly within the shell and have end plugs with drilled holes to permit draining drying and inerting with helium. The shell has holes in the top and bottom fittings to permit draining, drying and inerting of the assembly.

The total number of full length rods that can be placed in the Reconfigured Fuel Assembly is less than the number that are in the Yankee Class fuel assemblies (maximum of 64 versus 231 rods of the most reactive fuel). Consequently, the effects of a Reconfigured Fuel Assembly placed in a canister (e.g., criticality, thermal output, source term) are significantly less than the effects of a design basis Yankee Class spent fuel assembly. These effects are evaluated in the appropriate chapters that follow.

Figure 1.2-1 Vertical Concrete Storage Cask

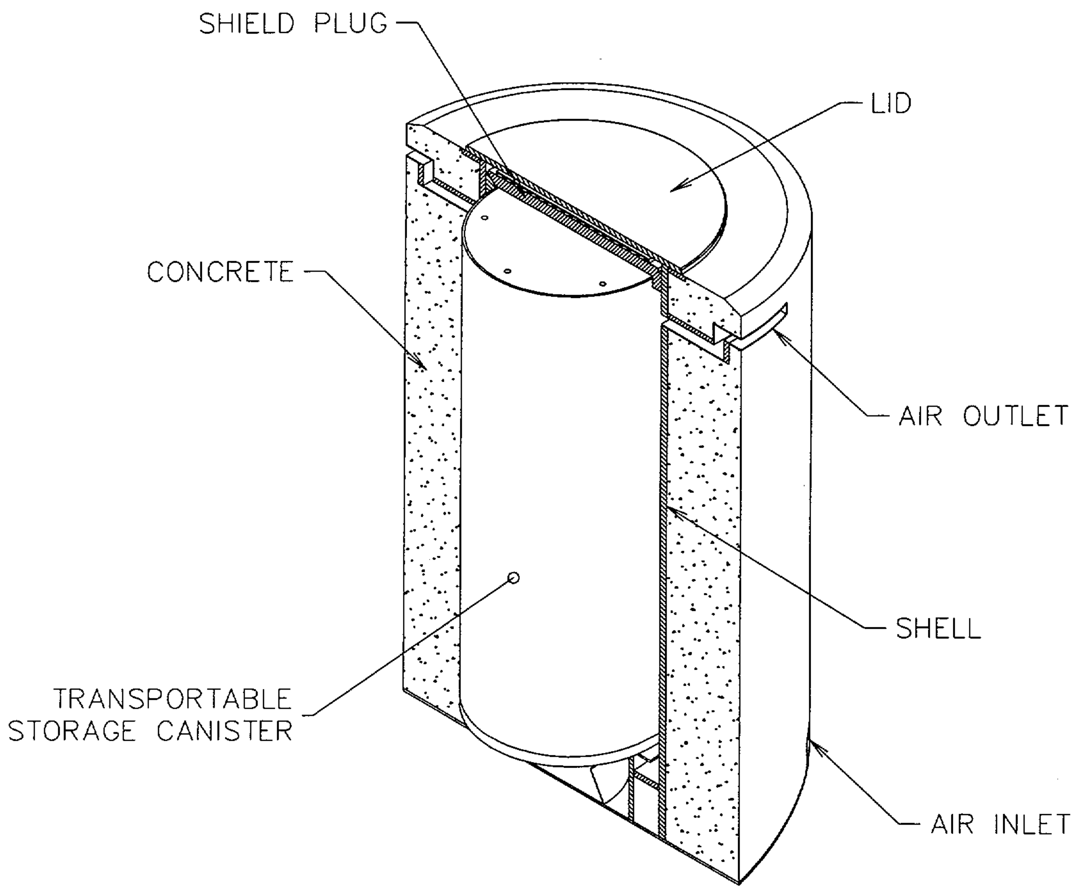

Figure 1.2-2 Transfer Cask

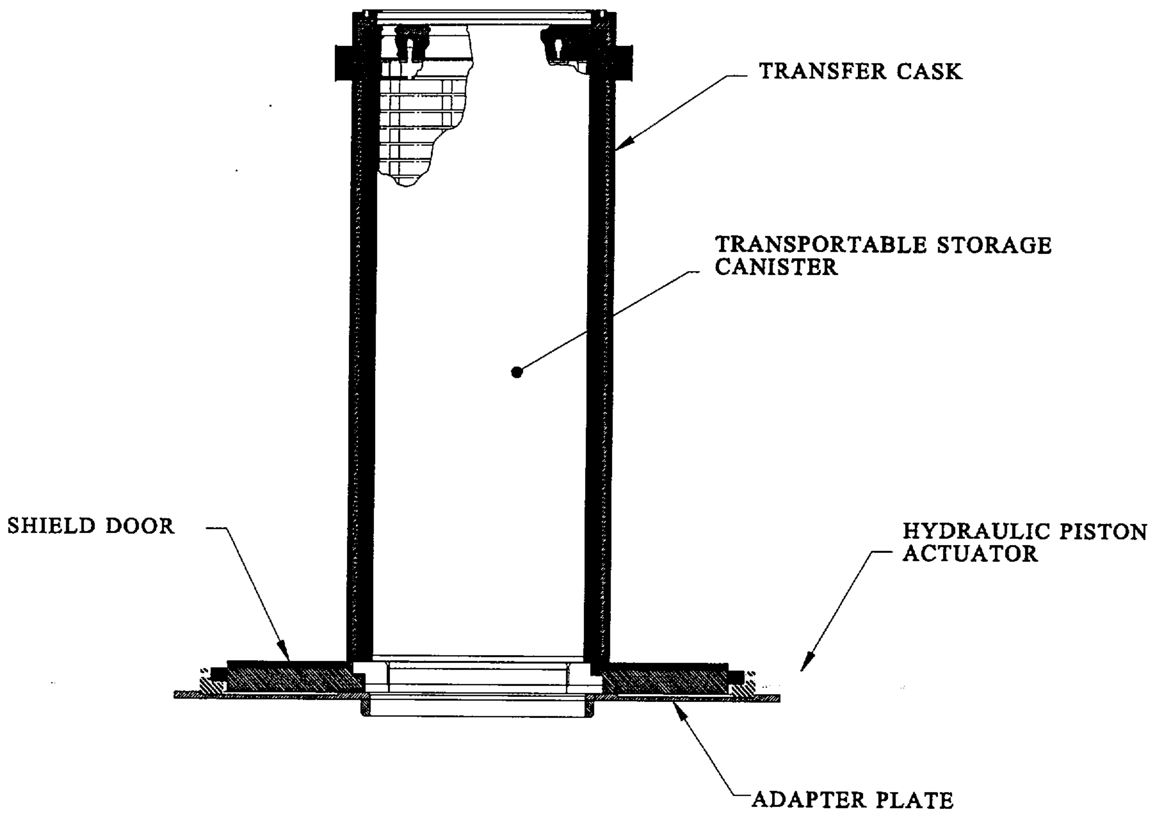

Figure 1.2-3 NAC-STC Transport Configuration

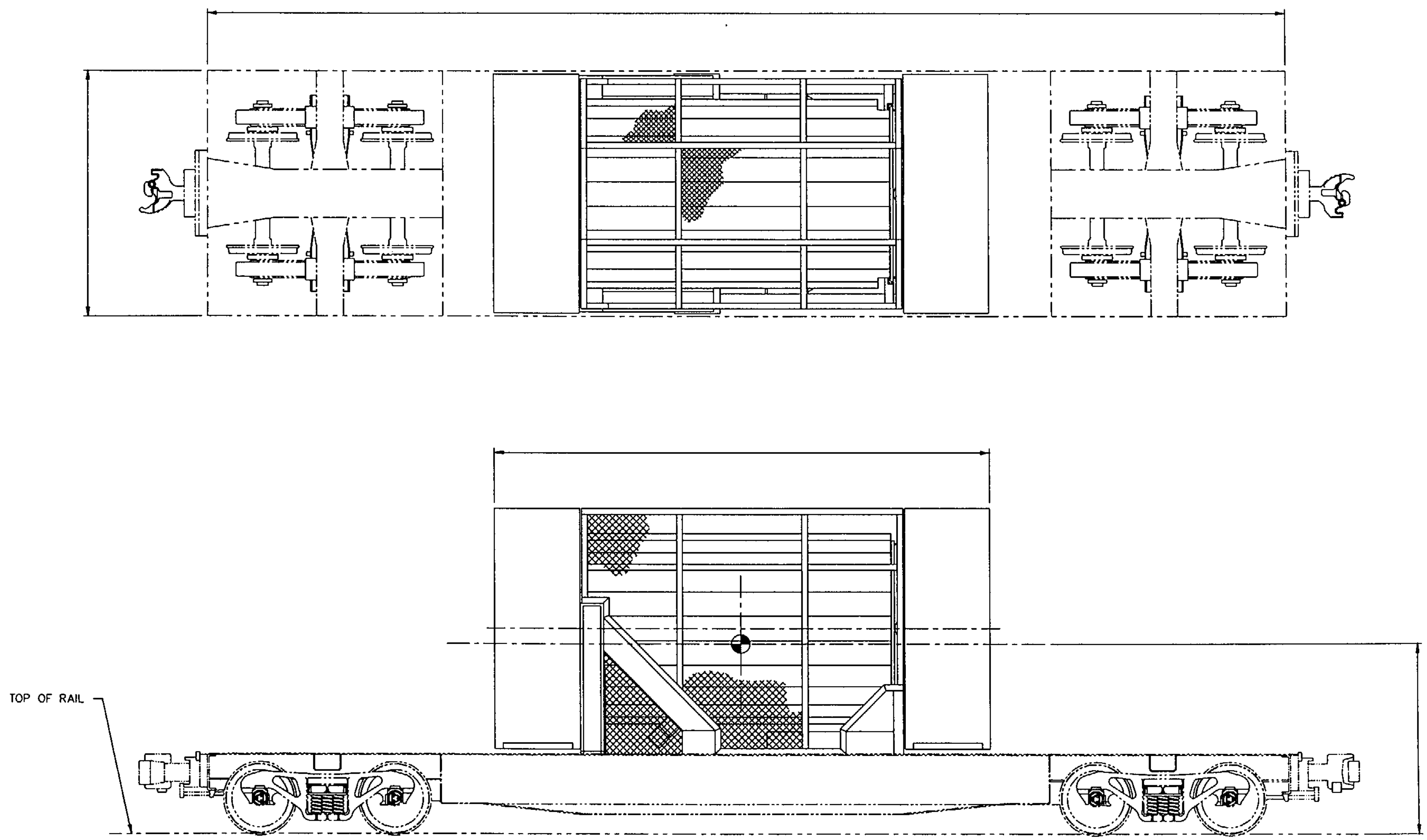

Figure 1.2-4 Reconfigured Fuel Assembly

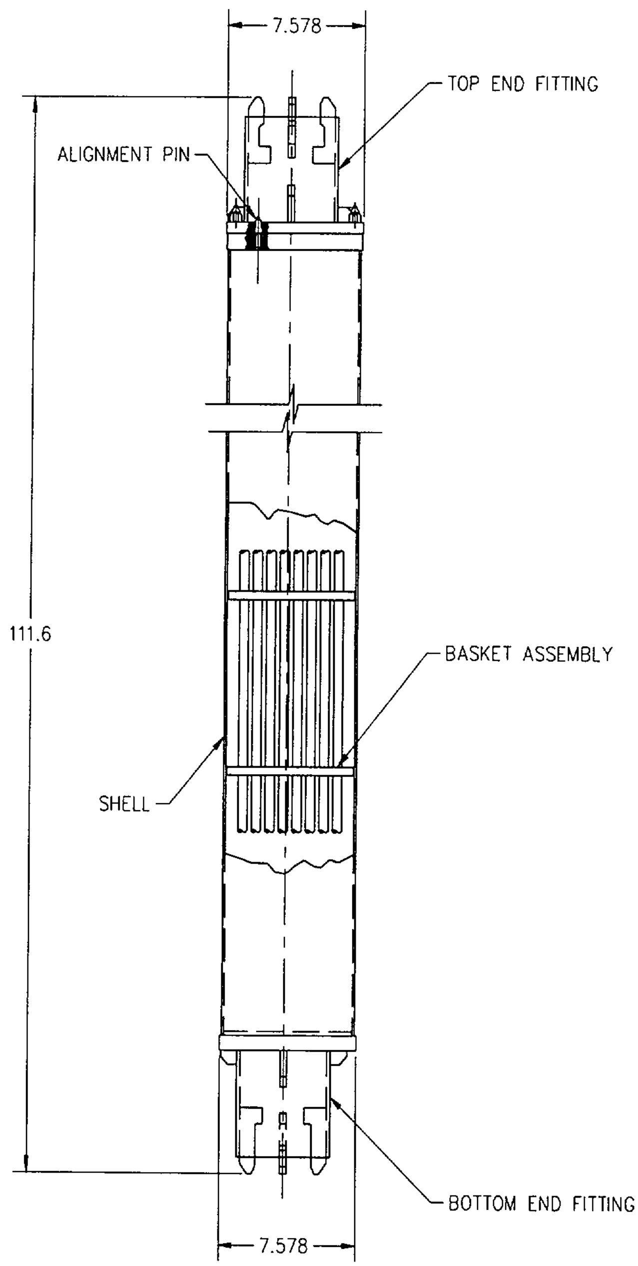

Figure 1.2-5 Transfer Cask and Canister Arrangement

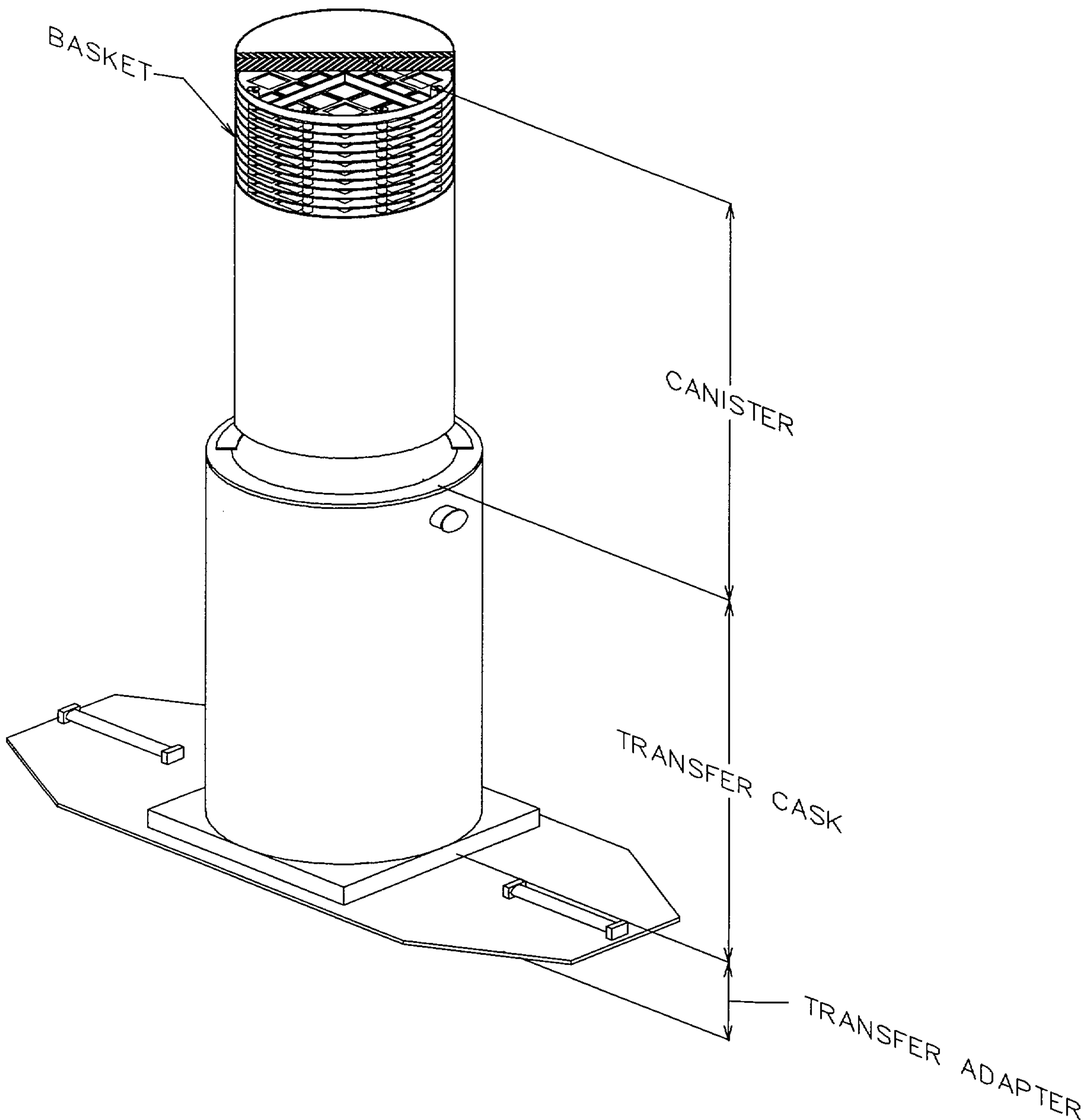

Figure 1.2-6 Vertical Concrete Cask and Transfer Cask Arrangement

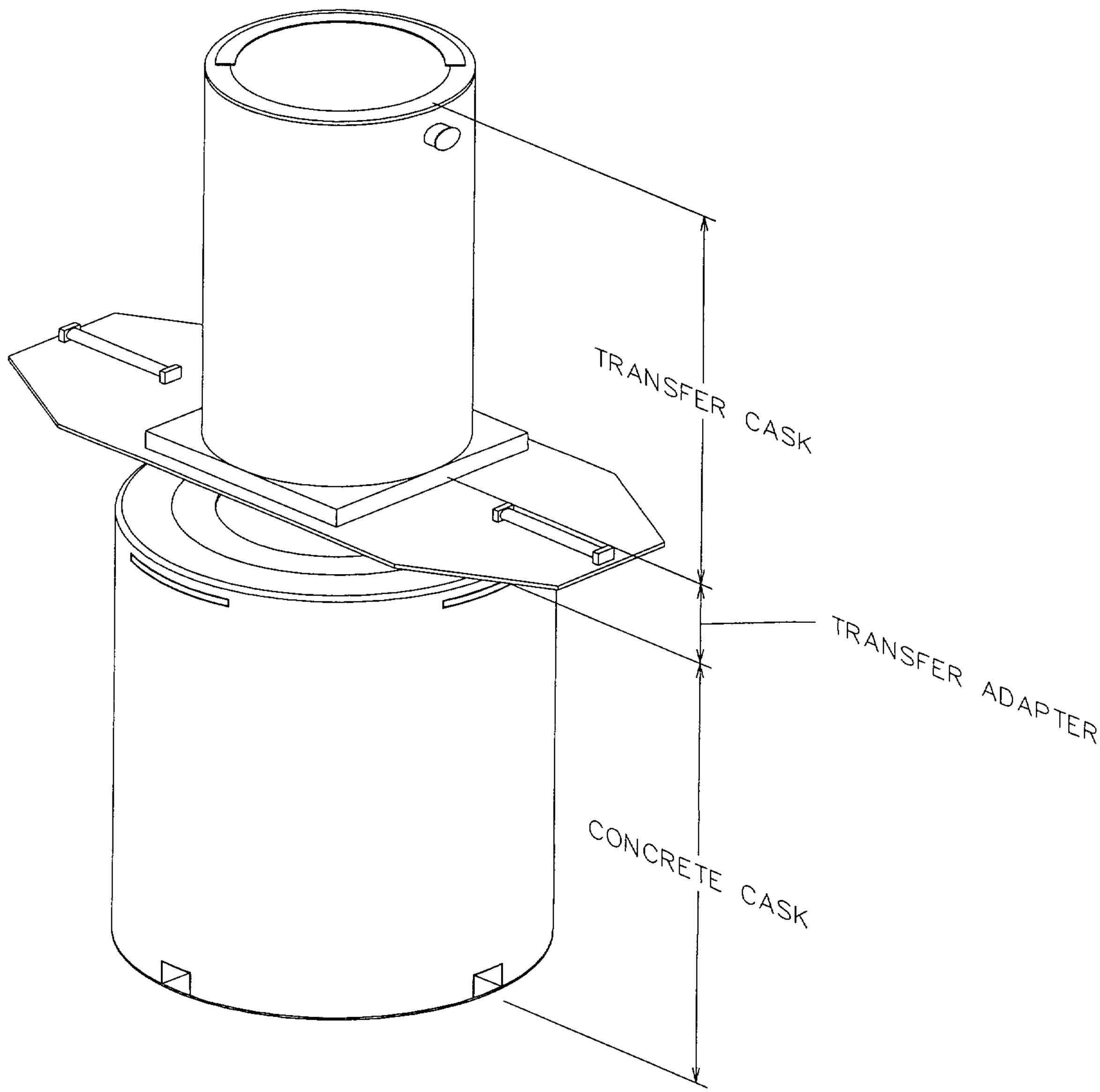

Figure 1.2-7 Major Component Configuration for Loading the Vertical Concrete Cask

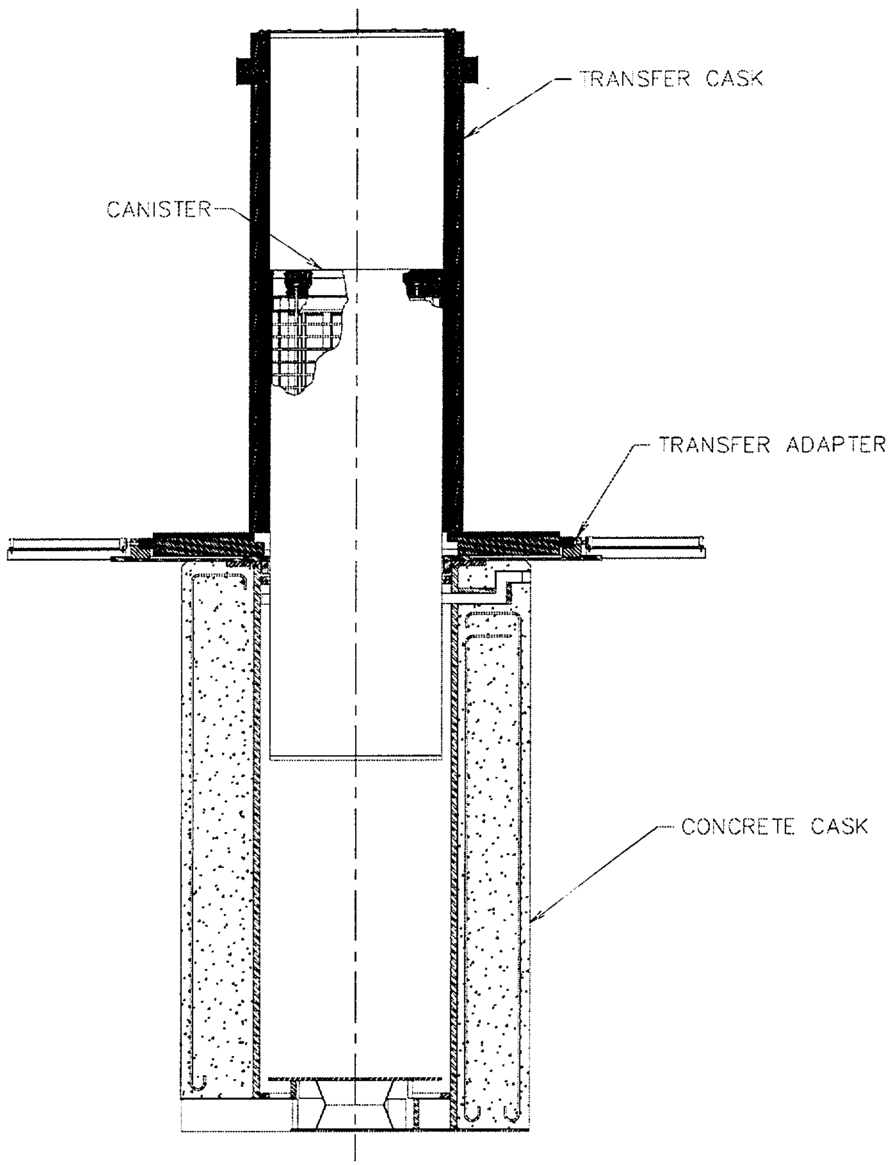

Table 1.2-1 Major Physical Design Parameters for the Transportable Storage Canister

Parameter	Design Value
Outside Diameter	70.64 in.
Length	122.5 in.
Capacity	36 Yankee Class spent fuel assemblies
Weight	54,730 lbs. (nominal) w/ fuel
Maximum heat load	12.5 kW (fuel)
Maximum fuel clad temperature Normal conditions Off-normal and accident	Zircaloy or Stainless Clad 340°C 430°C
Internal atmosphere	Helium

Table 1.2-2 Transportable Storage Canister Fabrication Specification Summary

Materials

- All material shall be in accordance with the referenced drawings and meet the applicable ASME standard.

Welding

- All welds shall be in accordance with the referenced drawings.
- All filler metals shall be appropriate ASME material.
- All welders and welding operators shall be qualified in accordance with ASME Section IX.
- All welding procedures shall be written and qualified in accordance with ASME Section IX.
- All welds specified to be visually examined shall be examined as specified in ASME Section V, Article 9 with acceptance per ASME Section III, NB-4424 and NB-4427.
- All welds specified to be liquid penetrant examined shall be examined in accordance with the requirements of ASME Section V, Article 6, with acceptance in accordance with ASME Section III, NB-5350.
- All personnel performing examinations shall be qualified in accordance with the NAC International quality assurance program and SNT-TC-1A.
- All welds specified to be radiographed shall be examined in accordance with the requirements of ASME Section V, Article 2, with acceptance per ASME Section III, NB 5320.
- All welds specified to be ultrasonically examined shall be examined in accordance with ASME Section V, Article 5, with acceptance in accordance with ASME Section III, NB-5330.

Fabrication

- All cutting, welding, and forming shall be in accordance with ASME, Section III, NB-4000 unless otherwise specified. Code stamping is not required.
- All surfaces shall be cleaned to a surface cleanness classification C or better as defined in ANSI N45.2.1, Section 2.
- All fabrication tolerances shall meet the requirements of the referenced drawings after fabrication.

Packaging

- Packaging and shipping shall be in accordance with ANSI N45.2.2.

Quality Assurance

- The canister shall be fabricated under a quality assurance program that meets 10 CFR 72 Subpart G and 10 CFR 71 Subpart H.
- The supplier's quality assurance program must be accepted by the licensee prior to initiation of work.
- Hold points for inspection of a completed basket assembly are verification of the basket assembly diameter and length, insertion of a "dummy" fuel assembly into each fuel tube, and insertion of the basket into the canister shell.

A Certificate of Compliance shall be issued by the fabricator stating that the canister meets the specifications and drawings.

Table 1.2-3 Major Physical Design Parameters for the Vertical Concrete Cask

Parameter	**Value**
Height	160 in.
Outside diameter	128 in.
Shielding (side wall) Concrete thickness Steel thickness	 21 in. 3.50 in.
Radiation dose rate (average): Side surface Top surface Air inlet/ outlet vents	 ≤ 50 mrem/hr ≤ 35 mrem/hr ≤ 100 mrem/hr
Weight	155,000 lbs. (nominal)
Air flow at design heat load	1 (lbs.-m)/sec
Material of construction Concrete Reinforcing steel Steel liner	 Type II Portland Cement A615 Grade 60 A36 Carbon Steel
Service life	50 years
Maximum concrete temperatures for normal operation	150°F bulk 200°F local

Table 1.2-4 Concrete Cask Fabrication Specification Summary

Materials

- Concrete mix shall be in accordance with the requirements of ACI 318 and ASTM C94.
- Type II Portland Cement, ASTM C150.
- Fine aggregate ASTM C33 and C637.
- Coarse aggregate ASTM C33 and C637.
- Admixtures

  Water Reducing ASTM C494.

  Pozzolanic Admixture ASTM C618.
- Compressive Strength 4000 psi at 28 days.
- Specified Air Entrainment 3% - 6%.
- All steel components shall be of material as specified in the referenced drawings.

Welding

- Visual inspection of all welds shall be performed to the requirements of AWS D1.1, Section 8.15.

Construction

- Specimens shall be obtained or prepared for each batch or truck load of concrete per ASTM C172 and ASTM C192.
- Test specimens shall be tested in accordance with ASTM C39.
- Formwork shall be in accordance with ACI 318.
- All sidewall formwork and shoring shall remain in place for at least 24 hours.
- All bottom formwork and shoring shall remain in place for 14 days.
- Grade, type, and details of all reinforcing steel shall be in accordance with the referenced drawings.
- Embedded items shall conform to ACI 318 and the referenced drawings.
- The placement of concrete shall be in accordance with ACI 318.
- Surface finish shall be in accordance with ACI 318.

Quality Assurance

- The concrete cask shall be constructed under a quality assurance program that meets 10 CFR 72 Subpart G. The quality assurance program must be accepted by NAC International and the licensee prior to initiation of the work.

Table 1.2-5 Major Physical Design Parameters for the Transfer Cask

Parameter	**Value**
Inside Diameter	71.5 in.
Outside Diameter	86.5 in.
Height	133.38 in.
Empty Weight (nominal)	80,800 lbs.
Side Wall Dose Rate (average)	≤ 200 mrem/hr

Table 1.2-6 NAC-MPC Design Basis Fuel Characteristics

Parameter	Yankee Class Spent Fuel[1, 2, 3,4]		
	United Nuclear Type A	Combustion Type A	Westinghouse Type B
Number of Assemblies per Canister[6]	36	36	34
Assembly Weight, lbs.	850	850	900
Assembly Length, in.	111.25	111.79	111.25
Active Fuel Length, in.	91	91	92
Fuel Rod Cladding	Zircaloy	Zircaloy	Stainless Steel
Maximum Uranium, kgU	245.6	239.4	286.9
Maximum Initial ^{235}U, wt %	4.0	3.9	4.94
Minimum Initial ^{235}U, wt %	4.0	3.7[5]	4.94
Maximum Burnup, MWD/MTU	32,000	36,000[5]	32,000
Maximum Assembly Decay Heat, kW	0.347	0.347	0.347
Maximum Decay Heat, kW	9.3	12.5	9.0
Minimum Cool Time, yr	13.0	8.1[5]	21.0

1. The Yankee Class spent fuel includes United Nuclear Type A and Type B, Combustion Engineering Type A and Type B, Exxon-ANF Type A and Type B, Westinghouse Type A and Type B. The United Nuclear Type A is the most reactive assembly and is used as the design basis fuel for criticality analyses. The Combustion Type A is the design basis fuel for shielding and thermal evaluations. The Westinghouse Type B fuel is the heaviest assembly and is the design basis fuel for structural considerations.

2. The Exxon-ANF fuel was provided by Exxon after Exxon was acquired by Advanced Nuclear Fuel (ANF). Fuel provided by ANF was designated Exxon-ANF. This fuel is considered to be Exxon fuel throughout this report. Except for lower enrichment and fuel hardware, Exxon fuel (and therefore, Exxon-ANF fuel) has the same characteristics as Combustion Engineering fuel.

3. The NAC-MPC can accommodate one or more Reconfigured Fuel Assemblies containing up to 64 fuel rods or rod segments classified as failed.

4. Exxon fuel assemblies with 3.5 wt % ^{235}U and burnups of 36,000 MWD/MTU require a minimum cool time of 9 or 16 years for Zircaloy or stainless steel fuel assemblies, respectively.

5. Combustion Engineering fuel assemblies with an initial minimum enrichment of 3.5 wt %, burnups up to 32,000 MWD/MTU require minimum cooling times of 8.0 years.

6. Up to 36 intact fuel assemblies of any type not exceeding 30,600 pounds total weight are authorized.

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1.3 Agents and Contractors

The prime contractor for the NAC-MPC design is NAC International Inc. (NAC). All design and specification activities are performed by NAC. Fabrication of the steel components will be by qualified vendors. A qualified concrete contractor will perform fabrication of the vertical concrete storage cask. All fabrication activities will be performed in accordance with quality assurance programs meeting the requirements of 10 CFR 71 and 10 CFR 72.

NAC is a private corporation founded in 1968, whose primary focus is the tracking, inspection, handling, storage, and transportation of spent nuclear fuel. NAC is recognized in the industry as expert in all aspects of the design, licensing, and operation of spent fuel handling, inspection, storage, and transport equipment, as well as in the management of spent fuel inventories.

NAC is the leading U.S. company in the transport and storage of spent nuclear fuel, owning and operating the largest fleet of commercial spent fuel transport casks in the United States. This fleet includes the following casks:

- 5 NLI-1/2 (truck) - 1 PWR/2 BWR - Approved for the transport of LWR and metallic fuel and for high level waste.
- 5 NAC LWT - 1 PWR/2 BWR - Approved for the transport of LWR fuel, metallic fuel, research reactor fuel and high level waste.
- 2 NLI-10/24 (Rail) - 10 PWR/24 BWR - Approved for the transport of LWR fuel.

These casks are approved by the U.S. NRC under 10 CFR 71 and have successfully and safely completed more than 1,000 shipments of spent fuel and high level waste for more than 40 nuclear facilities in the last 15 years. NAC has also designed and licensed the NAC-STC rail cask for the storage and transport of spent fuel. The NAC-MPC canister is designed to be transported in the NAC-STC.

NAC has designed, analyzed, and obtained NRC approval for the following dry storage casks:

- NAC-I26 S/T - Certificate of Compliance approval for storage of 26 PWR fuel assemblies (Docket 72-1002)
- NAC-I28 S/T - Site specific approval for the storage of 28 PWR fuel assemblies (Docket 72-1020)
- NAC-C28 S/T - Certificate of Compliance approval for the storage of 28 consolidated PWR fuel canisters (56 assemblies) (Docket 72-1003)

Within the last 10 years, NAC contractors have completed the fabrication of two NAC-I28 S/T and one NAC-I26 S/T storage casks, and five NAC-LWT transport casks.

NAC has also designed the NAC-STC rail cask for the storage and transport of directly loaded spent fuel. The NAC-STC is approved for use in site specific applications (Docket 72-1013) for storage and for transport (Certificate of Compliance Number 71-9235). NAC has also prepared, and submitted, a Safety Analysis Report for the transport of canistered Yankee Class spent fuel (Docket 71-9235).

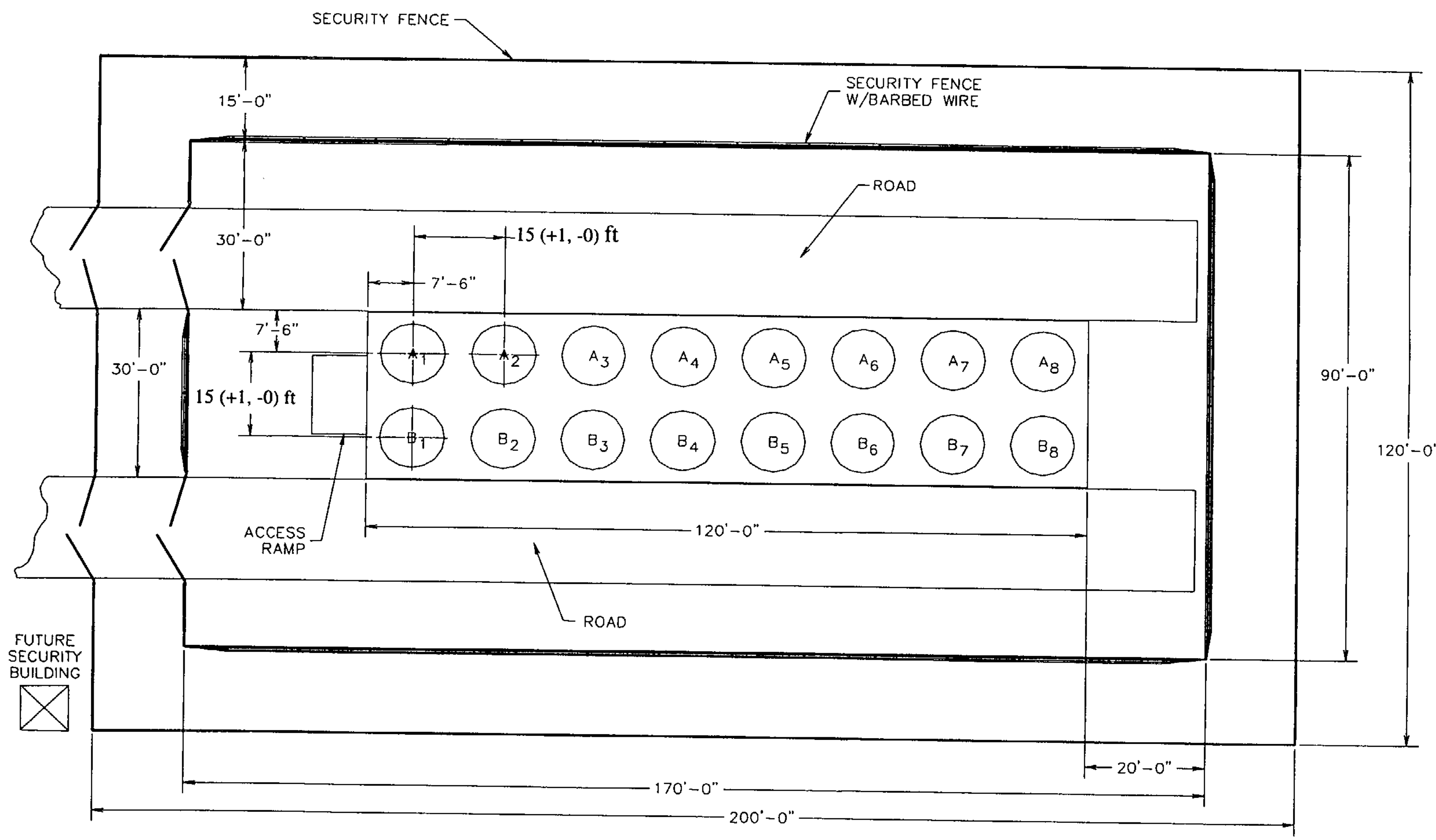
Safety Analysis Reports have also been prepared and submitted for the NAC Universal MPC System (NAC-UMS™) for the transport (Docket 71-9270) and long-term storage of several classes of spent fuel (Docket 72-1015).

1.4 Generic Storage Cask Arrays

A typical ISFSI storage pad layout for 16 storage casks is provided in Figure 1.4-1. As shown in this figure, roads parallel the sides of the pad to facilitate transfer of the storage cask from the transporter to the designated storage position on the pad. Loaded storage casks are placed in the vertical position on the pad in a linear array. Array sizes could accommodate from 1 to more than 200 casks. Figure 1.4-1 shows typical spacing and representative site dimensions. However, these are dependent on the general site layout, access roads, site boundaries, and transfer equipment selection.

The reinforced concrete foundation is capable of sustaining the transient loads from the air pad and the general loads of the stored casks. If necessary, the pad can be constructed in phases to specifically meet utility-required expansions.

Figure 1.4-1 Typical ISFSI Storage Pad Layout



1.5 License Drawings

This section presents the License Drawings for the NAC-MPC System.

1.5.1 NAC-MPC License Drawings

Drawing Number	Title	Revision No.	No. of Sheets
455-821	Adapter Ring, Transfer Adapter to NAC-STC MPC-Yankee	0	1
455-856	Name Plate - NAC-VCC Cask	0	1
455-859	Assembly, Transfer Adapter, MPC-Yankee	1	3
455-860	Assembly, Transfer Cask (TFR), MPC - Yankee	4	4
455-861	Weldment, Structure, Vertical Concrete Cask (VCC), MPC - Yankee	4	2
455-862	Loaded Vertical Concrete Cask (VCC), MPC - Yankee	2	1
455-863	Lid, Vertical Concrete Cask (VCC), MPC- Yankee	2	1
455-864	Shield Plug, Vertical Concrete Cask (VCC), MPC- Yankee	1	1
455-866	Reinforcing Bar and Concrete Placement, Vertical Concrete Cask (VCC), MPC- Yankee	0	3
455-870	Canister Shell, MPC- Yankee	3	1
455-871	Details, Canister, MPC- Yankee	4	2
455-872	Assembly, Transportable Storage Canister (TSC), MPC- Yankee	6	2
455-873	Assembly, Drain Tube, Canister, MPC- Yankee	2	1
455-881	PWR Fuel Tube, MPC- Yankee	3	1
455-891	Bottom Weldment, Fuel Basket, MPC- Yankee	0	1
455-892	Top Weldment, Fuel Basket, MPC- Yankee	1	1
455-893	Support Disk and Misc. Basket Details, MPC- Yankee	3	1
455-894	Heat Transfer Disk, Fuel Basket, MPC- Yankee	1	1
455-895	Fuel Basket Assembly, MPC- Yankee	2	1

1.5.2 Yankee Class Reconfigured Fuel Assembly License Drawings

Drawing Number	Title	Revision No.	No. of Sheets
YR-00-060	Yankee-Class Reconfigured Fuel Assembly	1	1
YR-00-061	Yankee-Class Reconfigured Fuel Assembly, Shell Weldment	1	1
YR-00-062	Yankee-Class Reconfigured Fuel Assembly, Top End Fitting Assembly	1	1
YR-00-063	Yankee-Class Reconfigured Fuel Assembly, Bottom End Fitting Assembly	1	1
YR-00-064	Yankee-Class Reconfigured Fuel Assembly, Nozzle Bolt and Alignment Pin	1	1
YR-00-065	Yankee-Class Reconfigured Fuel Assembly, Fuel Basket Assembly	1	1
YR-00-066	Yankee-Class Reconfigured Fuel Assembly, Fuel Tube Assembly	1	1

FIGURE WITHHELD UNDER 10 CFR 2.390

ASSY	ASSY	ASSY
QUANTITY		

SYM.	GEOMETRY
	FLATNESS
	STRAIGHTNESS
	ANGULARITY
	PERPENDICULARITY
	PARALLELISM
	CONCENTRICITY
	TRUE POSITION

TOLERANCES UNLESS OTHERWISE SPECIFIED

3 PLACE DEC.	TOL.	2 PLACE DEC.	TOL.
UNDER 3		UNDER 6	±.04
3-12		6-15	±.06
OVER 12		OVER 18	±.08
1 PLACE DEC.	±.1	FILLETS .03	
FINISH (√)-AA-125 MICRO IN		ANGLES ±.5°	
THREADS-UNIFIED CL2A-2B		CORNERS R.03	

NEXT ASSEMBLY:

DRAWING TYPE: LICENSE

GROUP	NAME	DATE
PREPARER		
CHECKER		
PROJECT MANAGER		
VP/CHIEF ENGINEER		
DIRECTOR of PROJECTS		
LICENSING MANAGER		
QUALITY		
BRU VP		

NAC INTERNATIONAL

ADAPTER RING,
TRANSFER ADAPTER
TO NAC-STC
MPC-YANKEE

PROJECT 455	DESIGN PACKAGE	DRAWING 821	REV 0
SCALE 1/6	EST.WT. 1725#	SH 1 OF 1	10:49AM 4-23-97

A

FIGURE WITHHELD UNDER 10 CFR 2.390

ASSY	ASSY	ASSY						NAC INTERNATIONAL
QUANTITY								
SYM	GEOMETRY	TOLERANCES UNLESS OTHERWISE SPECIFIED				GROUP	NAME	DATE
	FLATNESS	3 PLACE DEC.	TOL.	2 PLACE DEC.	TOL.	PREPARER	[illegible]	4/24/97
	STRAIGHTNESS	UNDER 3		UNDER 6	±.04	CHECKER	[illegible]	[illegible]
	ANGULARITY	3-12		6-18	±.06	PROJECT MANAGER	[illegible]	[illegible]
		OVER 12		OVER 18	±.08	VP/CHIEF ENGINEER	[illegible]	4/24/97
	PERPENDICULARITY	1 PLACE DEC.	±.1	FILLETS .03		DIRECTOR of PROJECTS	[illegible]	[illegible]
		FINISH (√)-AA-125 MICRO IN		ANGLES ±.5°		LICENSING MANAGER	[illegible]	[illegible]
	PARALLELISM	THREADS-UNIFIED CL2A-2B		CORNERS R.03				
	CONCENTRICITY	NEXT ASSEMBLY: 455-866				QUALITY	[illegible]	[illegible]
	TRUE POSITION	DRAWING TYPE: LICENSE				SR VP	[illegible]	[illegible]

NAMEPLATE- NAC-VCC CASK			
PROJECT 455	DESIGN PACKAGE	DRAWING 856	REV 0
SCALE FULL	EST.WT.	SH 1 OF 1	11:30AM 4-24-97

FIGURE WITHHELD UNDER 10 CFR 2.390

ASSY	ASSY	ASSY		
QUANTITY				

SYM.	GEOMETRY	TOLERANCES UNLESS OTHERWISE SPECIFIED				GROUP	NAME	DATE
	FLATNESS	3 PLACE DEC.	TOL.	2 PLACE DEC	TOL.	PREPARER	[illegible]	1-4-99
	STRAIGHTNESS	UNDER 3		UNDER 6	±.04	CHECKER	[illegible]	[illegible]
	ANGULARITY	3-12		6-18	±.06	PROJECT MANAGER	[illegible]	[illegible]
		OVER 12		OVER 18	±.09	VP/CHIEF ENGINEER	[illegible]	[illegible]
	PERPENDICULARITY	1 PLACE DEC.	±.1	FILLETS .03		DIRECTOR of PROJECTS	NA	
		FINISH (√)-AA-125 MICRO IN		ANGLES ±.5°		LICENSING MANAGER	[illegible]	[illegible]
	PARALLELISM	THREADS-UNIFIED CL2A-2B		CORNERS R.03		QUALITY	[illegible]	[illegible]
	CONCENTRICITY	NEXT ASSEMBLY:						
	TRUE POSITION	DRAWING TYPE: LICENSE				SBU VP	NA	

NAC INTERNATIONAL

ASSEMBLY,
TRANSFER ADAPTER,
MPC-YANKEE

PROJECT 455	DESIGN PACKAGE	DRAWING 859	REV 1
SCALE 1/10	EST. WT. NOTED	SH 1 OF 3	9:16AM 12-30-98

FIGURE WITHHELD UNDER 10 CFR 2.390

NAC INTERNATIONAL

ASSEMBLY,
TRANSFER ADAPTER,
MPC–YANKEE

PROJECT 455	DESIGN PACKAGE	DRAWING 859	REV 1
SCALE 1/10	EST.WT. NOTED	SH 2 OF 3	9:17AM 12-30-98

FIGURE WITHHELD UNDER 10 CFR 2.390

NAC INTERNATIONAL

ASSEMBLY,
TRANSFER ADAPTER,
MPC–YANKEE

PROJECT 455	DESIGN PACKAGE	DRAWING 859	REV 1
SCALE 1/5	EST.WT. NOTED	SH 3 OF 3	9:17AM 12-30-98

FIGURE WITHHELD UNDER 10 CFR 2.390

ASSY	ASSY	ASSY	ASSY	ASSY
QUANTITY				

DIMENSIONING AND TOLERANCING SHALL BE PER ANSI Y14.5-82
UNSPECIFIED DIMENSIONS AND TOLERANCES SHOWN BELOW.
DIMENSIONS ARE IN INCHES. FRACTIONAL TOLERANCE: ±1/8

SYM.	GEOMETRY	.XXX	TOL.	.XX	TOL.
	FLATNESS	UNDER 3		UNDER 6	±.04
	STRAIGHTNESS	3-12		6-18	±.06
	ANGULARITY	OVER 12		OVER 18	±.08
	PERPENDICULARITY	.X	±.1	ANGLES ±.5°	
	PARALLELISM	ALL UNSPECIFIED TOOL RADII .01 - .03			
	CONCENTRICITY	BREAK ALL SHARP CORNERS .01 - .03			
	TRUE POSITION	ALL UNSPECIFIED MACHINED SURFACES SHALL BE ▽ OR BETTER			

NEXT ASSEMBLY:

DRAWING TYPE: LICENSE

GROUP	NAME	DATE
PREPARER	[illegible]	[illegible]
CHECKED	[illegible]	[illegible]
PROJECT MANAGER	[illegible]	[illegible]
DIRECTOR DESIGN AND ANALYSIS	[illegible]	[illegible]
DIRECTOR LICENSING	[illegible]	[illegible]
VICE PRESIDENT, QUALITY	[illegible]	[illegible]

NAC INTERNATIONAL

ASSEMBLY,
TRANSFER CASK (TFR)
MPC-YANKEE

PROJECT 455 | DRAWING 860 | REV 4

SCALE 1/10 | EST. WT. | SH 1 OF 4

FIGURE WITHHELD UNDER 10 CFR 2.390

NAC INTERNATIONAL

ASSEMBLY,
TRANSFER CASK (TFR)
MPC–YANKEE

PROJECT 455		DRAWING 860	REV 4
SCALE 1/10	EST. WT.	SH 2 OF 4	

FIGURE WITHHELD UNDER 10 CFR 2.390

NAC INTERNATIONAL

ASSEMBLY,
TRANSFER CASK (TFR)
MPC-YANKEE

PROJECT 455		DRAWING 860	REV 4
SCALE 1/10	EST. WT.	SH 3 OF 4	

FIGURE WITHHELD UNDER 10 CFR 2.390

NAC INTERNATIONAL

ASSEMBLY,
TRANSFER CASK (TFR)
MPC–YANKEE

PROJECT 455		DRAWING 860	REV 4
SCALE 1/8	EST. WT.	SH 4 OF 4	

FIGURE WITHHELD UNDER 10 CFR 2.390

ASSY / ASSY / ASSY / ASSY / ASSY	QUANTITY	NAC INTERNATIONAL
DIMENSIONING AND TOLERANCING SHALL BE PER ANSI Y14.5-82. UNSPECIFIED DIMENSIONS AND TOLERANCES SHOWN BELOW. DIMENSIONS ARE IN INCHES. FRACTIONAL TOLERANCE: ±1/8	GROUP / NAME / DATE	WELDMENT, STRUCTURE, VERTICAL CONCRETE CASK (VCC) MPC-YANKEE
NEXT ASSEMBLY: 855-866		PROJECT 455
DRAWING TYPE: LICENSE		DRAWING 861 REV 4
		SCALE 1/16 EST WT SH 1 OF 2

FIGURE WITHHELD UNDER 10 CFR 2.390

NAC INTERNATIONAL

WELDMENT, STRUCTURE,
VERTICAL CONCRETE CASK (VCC)
MPC-YANKEE

PROJECT 455		DRAWING 861	REV 4
SCALE 1/16	EST WT.	SH 2 OF 2	

FIGURE WITHHELD UNDER 10 CFR 2.390

ASSY	ASSY	ASSY
QUANTITY		

SYM	GEOMETRY	TOLERANCES UNLESS OTHERWISE SPECIFIED				GROUP	NAME	DATE
	FLATNESS	3 PLACE DEC.	TOL.	2 PLACE DEC	TOL.	PREPARER	[illegible]	1-6-99
	STRAIGHTNESS	UNDER 3		UNDER 6	±.04	CHECKER	[illegible]	[illegible]
		3–12		6–18	±.06	PROJECT MANAGER	[illegible]	[illegible]
	ANGULARITY	OVER 12		OVER 18	±.09	VP/CHIEF ENGINEER	[illegible]	[illegible]
	PERPENDICULARITY	1 PLACE DEC.	±.1	FILLETS .03		DIRECTOR of PROJECTS	NA	
	PARALLELISM	FINISH (√)-AA-125 MICRO IN		ANGLES ±.5°		LICENSING MANAGER	[illegible]	[illegible]
	CONCENTRICITY	NEXT ASSEMBLY:				QUALITY	[illegible]	[illegible]
	TRUE POSITION	DRAWING TYPE: LICENSE				SBU VP	NA	

NAC INTERNATIONAL

LOADED VERTICAL CONCRETE CASK (VCC) MPC–YANKEE

PROJECT	DESIGN PACKAGE	DRAWING	REV
455		862	2
SCALE 1/8	EST. WT.	SH 1 OF 1	9:52AM 1-8-99

3 2 1

FIGURE WITHHELD UNDER 10 CFR 2.390

ASSY	ASSY	ASSY						
QUANTITY								

SYM.	GEOMETRY	TOLERANCES UNLESS OTHERWISE SPECIFIED				GROUP	NAME	DATE
	FLATNESS	3 PLACE DEC.	TOL.	2 PLACE DEC.	TOL.	PREPARER	R. [illegible]	1-4-99
	STRAIGHTNESS	UNDER 3		UNDER 6	±.04	CHECKER	[illegible]	1/6/99
		3–12		6–18	±.06	PROJECT MANAGER	[illegible]	1/6/99
	ANGULARITY	OVER 12		OVER 18	±.06	VP/CHIEF ENGINEER	[illegible]	1/6/99
	PERPENDICULARITY	1 PLACE DEC.	±.1	FILLETS .03		DIRECTOR of PROJECTS	NA	
	PARALLELISM	FINISH (√)-AA-125 MICRO IN		ANGLES ±.5°		LICENSING MANAGER	[illegible]	1/7/99
	CONCENTRICITY	NEXT ASSEMBLY:	455–862			QUALITY	RW Bass	1/7/99
	TRUE POSITION	DRAWING TYPE:	LICENSE			SBU VP	NA	

NAC INTERNATIONAL

LID,
VERTICAL CONCRETE CASK (VCC)
MPC–YANKEE

PROJECT 455	DESIGN PACKAGE	DRAWING 863	REV 2
SCALE 1/5	EST. WT.	SH 1 OF 1	1:28PM

A

FIGURE WITHHELD UNDER 10 CFR 2.390

ASSY	ASSY	ASSY
QUANTITY		

SYM	GEOMETRY	TOLERANCES UNLESS OTHERWISE SPECIFIED				GROUP	NAME	DATE
		3 PLACE DEC.	TOL.	2 PLACE DEC	TOL.			
	FLATNESS	UNDER 3		UNDER 6	±.04	PREPARER	[illegible]	1-4-99
	STRAIGHTNESS	3-12		6-18	±.06	CHECKER	[illegible]	1/6/99
	ANGULARITY	OVER 12		OVER 18	±.09	PROJECT MANAGER	[illegible]	1/6/99
	PERPENDICULARITY	1 PLACE DEC.	±.1	FILLETS .03		VP/CHIEF ENGINEER	[illegible]	1/6/99
	PARALLELISM	FINISH (√)-AA-125 MICRO IN		ANGLES ±.5°		DIRECTOR of PROJECTS	NA	
	CONCENTRICITY	THREADS-UNIFIED CL2A-2B		CORNERS R.03		LICENSING MANAGER	[illegible]	1/7/99
	TRUE POSITION	NEXT ASSEMBLY: 455-862				QUALITY	[illegible]	1/7/99
		DRAWING TYPE: LICENSE				BBU VP	NA	

NAC INTERNATIONAL

SHIELD PLUG,
VERTICAL CONCRETE CASK (VCC)
MPC-YANKEE

A

PROJECT 455	DESIGN PACKAGE	DRAWING 864	REV 1
SCALE 1/5	EST.WT.	SH 1 OF 1	1:30PM 1-4-99

FIGURE WITHHELD UNDER 10 CFR 2.390

ASSY	ASSY	ASSY
QUANTITY		

SYM	GEOMETRY
	FLATNESS
	STRAIGHTNESS
	ANGULARITY
	PERPENDICULARITY
	PARALLELISM
	CONCENTRICITY
	TRUE POSITION

TOLERANCES UNLESS OTHERWISE SPECIFIED			
3 PLACE DEC.	TOL.	2 PLACE DEC	TOL.
UNDER 3		UNDER 6	±.04
3-12		6-18	±.06
OVER 12		OVER 18	±.09
1 PLACE DEC.	±.1	FILLETS .03	
FINISH (√)-AA-125 MICRO IN		ANGLES ±.5°	
THREADS-UNIFIED CL2A-2B		CORNERS R.03	
NEXT ASSEMBLY: 455-862			
DRAWING TYPE: LICENSE			

GROUP		DATE
PREPARER	[illegible]	[illegible]
CHECKER	[illegible]	4/4/97
PROJECT MANAGER	[illegible]	[illegible]
VP/CHIEF ENGINEER	[illegible]	4/25/97
DIRECTOR OF PROJECTS	[illegible]	4/25/97
LICENSING MANAGER	[illegible]	[illegible]
QUALITY	[illegible]	[illegible]
BRU VP	[illegible]	4/25/97

NAC INTERNATIONAL

REINFORCING BAR AND CONCRETE PLACEMENT, VERTICAL CONCRETE CASK (VCC) MPC-YANKEE

PROJECT 455	DESIGN PACKAGE	DRAWING 866	REV 0
SCALE 1/16	EST. WT.	SH 1 OF 3	9:13AM 4-25-97

A

FIGURE WITHHELD UNDER 10 CFR 2.390

NAC INTERNATIONAL

REINFORCING BAR AND CONCRETE PLACEMENT, VERTICAL CONCRETE CASK (VCC) MPC-YANKEE

PROJECT 455	DESIGN PACKAGE	DRAWING 866	REV 0
SCALE 1/16	EST.WT.	SH 2 OF 3	

FIGURE WITHHELD UNDER 10 CFR 2.390

NAC INTERNATIONAL

REINFORCING BAR AND
CONCRETE PLACEMENT,
VERTICAL CONCRETE CASK (VCC)
MPC-YANKEE

PROJECT 455	DESIGN PACKAGE	DRAWING 866	REV 0
SCALE 1/12	EST. WT.	SH 3 OF 3	

FIGURE WITHHELD UNDER 10 CFR 2.390

GEOMETRY	TOLERANCES UNLESS OTHERWISE SPECIFIED				GROUP	NAME	DATE
FLATNESS	3 PLACE DEC.	TOL.	2 PLACE DEC	TOL.	PREPARER	[illegible]	[illegible]
STRAIGHTNESS	UNDER 3		UNDER 6	±.04	CHECKER	M. Luis	8-7-98
	3-12		6-18	±.06	PROJECT MANAGER	[illegible]	8-7-98
ANGULARITY	OVER 12		OVER 18	±.06	VP/CHIEF ENGINEER	N/A	
PERPENDICULARITY	1 PLACE DEC.	±.1	FILLETS .03		DIRECTOR PROJECTS	[illegible]	8-7-98
PARALLELISM	FINISH (√)-AA-125 MICRO IN		ANGLES ±.5°		LICENSING MANAGER	[illegible]	[illegible]
CONCENTRICITY	NEXT ASSEMBLY: 455-872				QUALITY	[illegible]	[illegible]
TRUE POSITION	DRAWING TYPE: LICENSE				SR VP	N/A	

NAC INTERNATIONAL

CANISTER SHELL,
MPC-YANKEE

PROJECT	DESIGN PACKAGE	DRAWING	REV
455		870	3
SCALE 1/8	EST. WT.	SH 1 OF 1	

FIGURE WITHHELD UNDER 10 CFR 2.390

95 ASSY	96 ASSY	97 ASSY	98 ASSY	99 ASSY	ITEM	NAME	MATERIAL	SPEC	DRAWING No.	DESCRIPTION
					1	LID SUPPORT RING	304 ST.STL.	ASTM A479		1/2 X 1/2 SQ. BAR

QUANTITY

DIMENSIONING AND TOLERANCING SHALL BE PER ANSI Y14.5-92 UNSPECIFIED DIMENSIONS AND TOLERANCES SHOWN BELOW. DIMENSIONS ARE IN INCHES. FRACTIONAL TOLERANCE: ±1/8

SYM	GEOMETRY	.XXX	TOL.	.XX	TOL.
	FLATNESS	UNDER 3		UNDER 6	±.04
	STRAIGHTNESS	3-12		6-18	±.06
	ANGULARITY	OVER 12		OVER 18	±.09
	PERPENDICULARITY	.X	±.1	ANGLES ±.5°	
	PARALLELISM	ALL UNSPECIFIED TOOL RADII: .01 – .03			
	CONCENTRICITY	BREAK ALL SHARP CORNERS .01 – .03			
	TRUE POSITION	ALL UNSPECIFIED MACHINED SURFACES SHALL BE ∇ OR BETTER			

NEXT ASSEMBLY: 455-872

DRAWING TYPE: LICENSE

GROUP	NAME	DATE
PREPARED	R. [signature]	11-23-99
CHECKED	[signature]	11/23/99
PROJECT ENGINEER	[signature]	11/23/99
STRUCTURAL DESIGN AND ANALYSIS	[signature]	11/23/99
SPECTRAL LICENSING	[signature]	11/23/99
VICE PRESIDENT QUALITY	[signature] for	11/24/99

NAC INTERNATIONAL

DETAILS,
CANISTER,
MPC-YANKEE

PROJECT 455 | DRAWING 871 | REV 4

SCALE 1/8 | EST. WT. | SH 1 OF 2 | 11-23-99

FIGURE WITHHELD UNDER 10 CFR 2.390

NAC INTERNATIONAL

DETAILS,
CANISTER,
MPC-YANKEE

PROJECT 455		DRAWING 871	REV 4
SCALE 1/8	EST WT	SH 2 OF 2	11-23-00

FIGURE WITHHELD UNDER 10 CFR 2.390

NAC INTERNATIONAL

ASSEMBLY, TRANSPORTABLE STORAGE CANISTER (TSC), MPC-YANKEE

DIMENSIONING AND TOLERANCING SHALL BE PER ANSI Y14.5-82
UNSPECIFIED DIMENSIONS AND TOLERANCES SHOWN BELOW
DIMENSIONS ARE IN INCHES. FRACTIONAL TOLERANCE: ±1/8

ALL UNSPECIFIED TOOL RADII: .01 - .03
BREAK ALL SHARP CORNERS .01 - .03

NEXT ASSEMBLY: 455-800/862

DRAWING TYPE: LICENSE

GROUP	NAME	DATE
PREPARED	R. Grallen	11-23-99
CHECKED	[illegible]	11/24/99
PROJECT MANAGER	[illegible]	11/24/99
DIRECTOR DESIGN AND ANALYSIS	W. [illegible] for TBD	11/29/99
DIRECTOR LICENSING	[illegible]	11/29/99
VICE PRESIDENT QUALITY	[illegible]	11/24/99

PROJECT 455 DRAWING 872 REV 6

SCALE 1/8 EST. WT. SH 1 OF 2

3

FIGURE WITHHELD UNDER 10 CFR 2.390

NAC INTERNATIONAL

ASSEMBLY, TRANSPORTABLE STORAGE CANISTER (TSC), MPC-YANKEE

PROJECT 455		DRAWING 872	REV 6
SCALE 1/8	EST.WT.	SH 2 OF 2	

FIGURE WITHHELD UNDER 10 CFR 2.390

ASSY	ASSY	ASSY
QUANTITY		

SYM	GEOMETRY
	FLATNESS
	STRAIGHTNESS
	ANGULARITY
	PERPENDICULARITY
	PARALLELISM
	CONCENTRICITY
	TRUE POSITION

TOLERANCES UNLESS OTHERWISE SPECIFIED			
3 PLACE DEC.	TOL.	2 PLACE DEC.	TOL.
UNDER 3		UNDER 6	±.04
3-12		6-18	±.06
OVER 12		OVER 18	±.09
1 PLACE DEC.	±.1	FILLETS .03	
FINISH (√)-AA-125 MICRO IN		ANGLES ±.5'	
THREADS-UNIFIED CL2A-2B		CORNERS R.03	
NEXT ASSEMBLY:	455-872		
DRAWING TYPE:	LICENSE		

GROUP	NAME	DATE
PREPARER	[illegible]	[illegible]
CHECKER	R. [illegible]	[illegible]
PROJECT MANAGER	[illegible]	[illegible]
VP/CHIEF ENGINEER	NA	
DIRECTOR of PROJECTS	[illegible]	[illegible]
LICENSING MANAGER	[illegible]	[illegible]
QUALITY	[illegible]	[illegible]
BU VP	NA	

NAC INTERNATIONAL

ASSEMBLY,
DRAIN TUBE,
CANISTER
MPC-YANKEE

PROJECT 455	DESIGN PACKAGE	DRAWING 873	REV 2
SCALE FULL	EST WT	SH 1 OF 1	1:45PM 5-13-98

A

FIGURE WITHHELD UNDER 10 CFR 2.390

NAC INTERNATIONAL

PWR FUEL TUBE,
MPC–YANKEE

NEXT ASSEMBLY: 455–895

DRAWING TYPE: LICENSE

PROJECT 455

DRAWING 881

REV 3

SCALE 1/1

SH 1 OF 1

FIGURE WITHHELD UNDER 10 CFR 2.390

ASSY	ASSY	ASSY
QUANTITY		

SYM	GEOMETRY
	FLATNESS
	STRAIGHTNESS
	ANGULARITY
	PERPENDICULARITY
	PARALLELISM
	CONCENTRICITY
	TRUE POSITION

TOLERANCES UNLESS OTHERWISE SPECIFIED			
3 PLACE DEC.	TOL.	2 PLACE DEC.	TOL.
UNDER 3		UNDER 6	±.04
3–12		6–18	±.06
OVER 12		OVER 18	±.09
1 PLACE DEC.	±.1	FILLETS .03	
FINISH (√)-AA-125 MICRO IN		ANGLES ±.5°	
THREADS-UNIFIED CL2A-2B			
RADIUS CORNERS .03			
TYPE OF DRAWING: LICENSE			

GROUP	NAME	DATE
DRAWN		
CHECK		
DESIGN AND ANALYSIS		
LICENSING		
PROJECTS		
QUALITY		
US ENGINEERING		
NEXT ASSEMBLY	455–895	

NAC INTERNATIONAL

BOTTOM WELDMENT,
FUEL BASKET,
MPC–YANKEE

A

PROJECT 455	DESIGN PACKAGE	DRAWING 891	REV 0
SCALE 1/5	EST. WT.	SH 1 OF 1	

FIGURE WITHHELD UNDER 10 CFR 2.390

ASSY	ASSY	ASSY		
QUANTITY				

SYM	GEOMETRY	TOLERANCES UNLESS OTHERWISE SPECIFIED				GROUP	NAME	DATE
	FLATNESS	3 PLACE DEC.	TOL.	2 PLACE DEC.	TOL.	PREPARER	[illegible]	[illegible]
	STRAIGHTNESS	UNDER 3		UNDER 6	±.04	CHECKER	[illegible]	[illegible]
		3-12		6-18	±.08	PROJECT MANAGER	[illegible]	[illegible]
	ANGULARITY	OVER 12		OVER 18	±.08	VP/CHIEF ENGINEER	[illegible]	[illegible]
	PERPENDICULARITY	1 PLACE DEC.	±.1	FILLETS .03		DIRECTOR of PROJECTS	[illegible]	[illegible]
	PARALLELISM	FINISH (√)-AA-125 MOD II		ANGLES ±.5°		LICENSING MANAGER	[illegible]	[illegible]
		THREADS-UNIFIED CL2A-2B		CORNERS R.03		QUALITY	[illegible]	[illegible]
	CONCENTRICITY	NEXT ASSEMBLY:	455-895					
	TRUE POSITION	DRAWING TYPE:	LICENSE			SR VP	[illegible]	[illegible]

NAC INTERNATIONAL

TOP WELDMENT,
FUEL BASKET,
MPC-YANKEE

PROJECT 455	DESIGN PACKAGE	DRAWING 892	REV 1
SCALE 1/5	EST.WT.	SH 1 OF 1	[illegible] 4-15-97

A

FIGURE WITHHELD UNDER 10 CFR 2.390

ASSY	ASSY	ASSY									NAC INTERNATIONAL	
QUANTITY												
SYM	GEOMETRY	TOLERANCES UNLESS OTHERWISE SPECIFIED				GROUP	NAME	DATE			SUPPORT DISK AND MISC. BASKET DETAILS, MPC–YANKEE	A
	FLATNESS	3 PLACE DEC	TOL.	2 PLACE DEC	TOL.	PREPARER	R. [illegible]	6-25-97				
	STRAIGHTNESS	UNDER 3		UNDER 6	±.04	CHECKER	[illegible]	6/25/97				
	ANGULARITY	3–12		6–18	±.06	PROJECT MANAGER	[illegible]	[illegible]				
		OVER 12		OVER 18	±.09	VP/CHIEF ENGINEER	[illegible]	6/25/97				
	PERPENDICULARITY	1 PLACE DEC.	±.1	FILLETS .03		DIRECTOR OF PROJECTS	NA					
	PARALLELISM	FINISH (√)–AA–125 MICRO IN		ANGLES ±.5°		LICENSING MANAGER	[illegible]	[illegible]				
		THREADS–UNIFIED CL2A–2B		CORNERS R.03								
	CONCENTRICITY	NEXT ASSEMBLY: 455–895				QUALITY	[illegible]	6/25/97	PROJECT 455	DESIGN PACKAGE	DRAWING 893	REV 3
	TRUE POSITION	DRAWING TYPE: LICENSE				BRU VP	NA		SCALE 1/5	EST.WT. NOTED	SH 1 OF 1	

FIGURE WITHHELD UNDER 10 CFR 2.390

ASSY	ASSY	ASSY	
QUANTITY			

SYM.	GEOMETRY
⏥	FLATNESS
—	STRAIGHTNESS
∠	ANGULARITY
⊥	PERPENDICULARITY
//	PARALLELISM
◎	CONCENTRICITY
⌖	TRUE POSITION

TOLERANCES UNLESS OTHERWISE SPECIFIED			
3 PLACE DEC.	TOL.	2 PLACE DEC.	TOL.
UNDER 3		UNDER 6	±.04
3–12		6–18	±.06
OVER 12		OVER 18	±.09
1 PLACE DEC.	±.1	FILLETS .03	
FINISH (√)-AA-125 MICRO IN		ANGLES ±.5°	
THREADS-UNIFIED CL2A-2B			
RADIUS CORNERS .03			
TYPE OF DRAWING:	LICENSE		

GROUP	NAME	DATE
DRAWN	[illegible]	[illegible]
CHECK	R. [illegible]	7-30-98
DESIGN AND ANALYSIS	[illegible]	[illegible]
LICENSING	[illegible]	[illegible]
PROJECTS	[illegible]	[illegible]
QUALITY	[illegible]	8/4/98
V.P. ENGINEERING	NA	
NEXT ASSEMBLY	455-895	

NAC INTERNATIONAL

HEAT TRANSFER DISK,
FUEL BASKET,
MPC–YANKEE

PROJECT 455	DESIGN PACKAGE	DRAWING 894	REV 1
SCALE 1/5	EST.WT. 57 LBS	SH 1 OF 1	2:39PM 7-30-98

FIGURE WITHHELD UNDER 10 CFR 2.390

NAC INTERNATIONAL

FUEL BASKET ASSEMBLY, MPC-YANKEE

NEXT ASSEMBLY: 455-872	PROJECT 455	DRAWING 895 REV 2
DRAWING TYPE: LICENSE	SCALE 1/6	SH 1 OF 1

A

FIGURE WITHHELD UNDER 10 CFR 2.390

ASSY	ASSY	ASSY
QUANTITY		

SYM.	GEOMETRY
⏥	FLATNESS
—	STRAIGHTNESS
∠	ANGULARITY
⊥	PERPENDICULARITY
//	PARALLELISM
◎	CONCENTRICITY
⌖	TRUE POSITION

TOLERANCES UNLESS OTHERWISE NOTED

SCALE: AS NOTED

YANKEE ATOMIC ELECTRIC COMPANY
580 MAIN STREET BOLTON, MA.

NUCLEAR SERVICES DIVISION

TITLE: YANKEE-CLASS RECONFIGURED FUEL ASSEMBLY

DWG. NO. YR-00-060

JOB NO. W.O. F500

FIGURE WITHHELD UNDER 10 CFR 2.390

ASSY	ASSY	ASSY		MATERIAL	SPEC	DRAWING NO.	DESCRIPTION
QUANTITY							

SYM.	GEOMETRY	TOLERANCES UNLESS OTHERWISE NOTED			
	FLATNESS	3 PLACE DEC.	TOL.	2 PLACE DEC.	TOL.
	STRAIGHTNESS	UNDER 3	±.003	UNDER 6	±.02
	ANGULARITY	3-12	±.005	6-18	±.03
	PERPENDICULARITY	OVER 12	±.010	OVER 18	±.06
	PARALLELISM	1 PLACE DEC.	±.1	FILLETS .015-.030	
	CONCENTRICITY	FINISH√-AA-125 MICRO IN		ANGLES ±.5°	
	TRUE POSITION	THREADS-UNIFIED CL2A-2B			
		RADIUS CORNERS .015-.030			
		SCALE: AS NOTED			

YANKEE ATOMIC ELECTRIC COMPANY
580 MAIN STREET BOLTON, MA.

NUCLEAR SERVICES DIVISION

TITLE: YANKEE-CLASS RECONFIGURED FUEL ASSEMBLY SHELL WELDMENT

DWG. NO. YR-00-061

JOB NO. W.O. F500

FIGURE WITHHELD UNDER 10 CFR 2.390

ASSY	ASSY	ASSY
QUANTITY		

SYM.	GEOMETRY
▱	FLATNESS
—	STRAIGHTNESS
∠	ANGULARITY
⊥	PERPENDICULARITY
//	PARALLELISM
◎	CONCENTRICITY
⌖	TRUE POSITION

TOLERANCES UNLESS OTHERWISE NOTED

3 PLACE DEC.	TOL.	2 PLACE DEC.	TOL.
UNDER 3	±.003	UNDER 6	±.02
3-12	±.005	6-18	±.03
OVER 12	±.010	OVER 18	±.06
1 PLACE DEC.	±.1	FILLETS .015-.030	
FINISH ✓-AA 125 MICRO IN		ANGLES ±.5°	
THREADS-UNIFIED CLASS-2B			
RADIUS CORNERS .015-.030			
SCALE:			

YANKEE ATOMIC ELECTRIC COMPANY
580 MAIN STREET BOLTON, MA.

NUCLEAR SERVICES DIVISION

TITLE: YANKEE-CLASS RECONFIGURED FUEL ASSEMBLY TOP END FITTING ASSEMBLY

DWG. NO. YR-00-062

JOB NO. W.O. F500

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FIGURE WITHHELD UNDER 10 CFR 2.390

YANKEE ATOMIC ELECTRIC COMPANY
580 MAIN STREET BOLTON, MA.
NUCLEAR SERVICES DIVISION

TITLE: YANKEE-CLASS RECONFIGURED FUEL ASSEMBLY BOTTOM END FITTING ASSY.

DWG. NO. YR-00-063

JOB NO. W.O. F500

SCALE:

FIGURE WITHHELD UNDER 10 CFR 2.390

ASSY	ASSY	ASSY
QUANTITY		

SYM.	GEOMETRY
⏥	FLATNESS
—	STRAIGHTNESS
∠	ANGULARITY
⊥	PERPENDICULARITY
//	PARALLELISM
◎	CONCENTRICITY
⌖	TRUE POSITION

TOLERANCES UNLESS OTHERWISE NOTED

3 PLACE DEC.	TOL.	2 PLACE DEC.	TOL.
UNDER 3	±.003	UNDER 6	±.02
3-12	±.005	6-18	±.03
OVER 12	±.010	OVER 18	±.06
1 PLACE DEC.	±.1	FILLETS .015-.030	
FINISH-√-AA-125 MICRO IN		ANGLES ±.5°	
THREADS-UNIFIED CLASS-2B			
MATING CORNERS .010-.030			
SCALE:			

YANKEE ATOMIC ELECTRIC COMPANY
580 MAIN STREET BOLTON, MA.

NUCLEAR SERVICES DIVISION

TITLE: YANKEE-CLASS RECONFIGURED FUEL ASSEMBLY NOZZLE BOLT AND ALIGNMENT PIN

DWG. NO. YR-00-064

JOB NO. W.O. F500

FIGURE WITHHELD UNDER 10 CFR 2.390

102	101	100	ITEM	NAME	MATERIAL	SPEC	DRAWING NO.	DESCRIPTION
ASSY	ASSY	ASSY						

QUANTITY

SYM.	GEOMETRY
▱	FLATNESS
—	STRAIGHTNESS
∠	ANGULARITY
⊥	PERPENDICULARITY
//	PARALLELISM
◎	CONCENTRICITY
⊕	TRUE POSITION

TOLERANCES UNLESS OTHERWISE NOTED

3 PLACE DEC.	TOL.	2 PLACE DEC.	TOL.
UNDER 3	±.003	UNDER 6	±.02
3-12	±.005	6-18	±.03
OVER 12	±.010	OVER 18	±.05
1 PLACE DEC.	±.1	FILLETS .015-.030	
FINISH √-AA-125 MICRO IN		ANGLES ±.5°	
THREADS-UNIFIED CL2A-2B			
RADIUS CORNERS .015-.030			
SCALE:			

YANKEE ATOMIC ELECTRIC COMPANY
580 MAIN STREET BOLTON, MA.

NUCLEAR SERVICES DIVISION

TITLE: YANKEE-CLASS
RECONFIGURED
FUEL ASSEMBLY
FUEL BASKET ASSEMBLY

DWG. NO. YR-00-065

JOB NO. W.O. F500

FIGURE WITHHELD UNDER 10 CFR 2.390

ASSY	ASSY	ASSY
QUANTITY		

SYM.	GEOMETRY
	FLATNESS
	STRAIGHTNESS
	ANGULARITY
	PERPENDICULARITY
	PARALLELISM
	CONCENTRICITY
	TRUE POSITION

TOLERANCES UNLESS OTHERWISE NOTED			
3 PLACE DEC.	TOL.	2 PLACE DEC.	TOL.
UNDER 3	±.003	UNDER 6	±.02
3-12	±.005	6-18	±.03
OVER 12	±.010	OVER 18	±.05
1 PLACE DEC.	±.1	FILLETS .015-.030	
[illegible]		ANGLES ±.5°	
[illegible]			
RADIUS CORNERS .010-.030			
SCALE:			

YANKEE ATOMIC ELECTRIC COMPANY
580 MAIN STREET BOLTON, MA.

NUCLEAR SERVICES DIVISION

TITLE: YANKEE-CLASS
RECONFIGURED
FUEL ASSEMBLY
FUEL TUBE ASSEMBLY

DWG. NO. YR-00-066

JOB NO. W.O. F500

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2.0 PRINCIPAL DESIGN CRITERIA

The NAC-MPC is a canister-based dry storage cask system that is designed to be transported in the NAC-STC licensed transport cask. It is designed to store Yankee Class spent fuel.

This chapter presents the design basis, including the principal design criteria, limiting load conditions, and operational parameters of the NAC-MPC dry storage system. The principal design criteria are summarized in Table 2-1.

Table 2-1 Summary of the NAC-MPC Design Criteria

Design Criteria	
Design Life	50 years
Design Code - Confinement	ASME Code, Section III, Subsection NB for confinement boundary
Design Code - Nonconfinement	
Basket	ASME Code, Section III, Subsection NG and NUREG/CR-6322
Vertical Concrete Cask	ACI-349, ACI-318, ANSI/ANS 57.9
Transfer Cask	ANSI N14.6 and NUREG-0612
Design Weight:	
Canister Assembly w/fuel	54,730 lbs.
Transfer Cask	80,743 lbs.
Vertical Concrete Cask	151,364 lbs.
Thermal:	
Maximum Temperature, Zircaloy Cladding	340°C for 10-yr. Cooled 380°C for 5-yr. Cooled 430°C Off-Normal/Accident/Transfer
Maximum Temperature, Stainless Steel Cladding	340°C for 10-yr. Cooled 430°C Off-Normal/Accident/Transfer
Ambient Temperature Range	-40° to 125°F
Average Annual Ambient Temperature	75°F
Concrete Temperature:	
Normal Conditions	≤ 150°F; ≤ 200°F local
Off-Normal/Accident Conditions	≤ 350°F local/ surface
Canister Cavity Atmosphere	Helium

Table 2-1 Summary of the NAC-MPC Design Criteria (Continued)

Design Criteria	
RADIATION PROTECTION/SHIELDING	
Concrete Cask Side Wall Contact Dose Rate	< 50 mrem/hr.
Concrete Cask Top Lid Contact Dose Rate	< 35 mrem/hr.
Concrete Cask Air Inlet/Outlet	< 100 mrem/hr.
Owner Controlled Area Boundary Normal/Off-Normal Annual Whole Body Dose Accident Whole Body Dose	 25 mrem/yr. 5 rem
SPENT FUEL SPECIFICATIONS	
Spent Fuel Type	Yankee Class
Fuel Configuration/Vendor	Westinghouse 18 x 18, 4.94 wt % ^{235}U United Nuclear 16 x 16, 4.0 wt % ^{235}U Combustion Engineering 16 x 16, 3.5 to 3.9 wt % ^{235}U Exxon 16 x 16, 3.5 to 4.0 wt % ^{235}U
Fuel Cladding	Stainless Steel - Westinghouse Zircaloy - All others
Spent Fuel Capacity – Intact Fuel Assemblies (May include one or more Reconfigured Fuel Assemblies)	36 United Nuclear Assemblies 36 Combustion Engineering Assemblies 36 Exxon Assemblies, or 34 Westinghouse Assemblies Up to 36 Fuel Assemblies of any Type Not Exceeding 30,600 pounds Total Weight
Spent Fuel Assembly Burnup (max)	36,000 MWD/MTU[1]
Decay Heat/Fuel Assembly or Reconfigured Fuel Assembly Zircaloy Clad Fuel Stainless Steel Clad Fuel Reconfigured Fuel Assembly	 0.347 kW 0.264 kW 0.102 kW

1. Based on the design basis, Combustion Engineering fuel at 36,000 MWD/MTU cooled 8.1 years. Exxon fuel is limited to 34,000 MWD/MTU and 10 years minimum cool time. The maximum burnup of all other fuel types is 32,000 MWD/MTU.

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2.1 Spent Fuel To Be Stored

The NAC-MPC has been designed to safely store up to 36 Yankee Class spent fuel assemblies. The spent fuel designs are delineated by various factors including manufacturer, type, enrichment, burnup, cool time, and cladding material. The design basis consists of two types, designated A and B. The Type A assembly incorporates a protruding corner of fuel pins while the Type B assembly omits one corner of the fuel rods. These fuel types, as well as minor differences among manufacturers, are illustrated in Figures 6.2-1 through 6.2-3. During reactor operations, the symmetric stacking of the alternating assemblies permitted the insertion of cruciform control blades between the assemblies. Table 2.1-1 lists the nominal design parameters of each fuel design type. These parameters, including the minimum enrichment limit, exclude the loading of unenriched fuel assemblies in the transportable storage canister.

2.1.1 Bounding Fuel Evaluation

The criticality evaluations show that the United Nuclear Type A 16 x 16 fuel assembly is the most reactive fuel type, even though the stainless steel clad Westinghouse fuel has a higher enrichment (4.94 wt % ^{235}U). The criticality evaluation considers a complete assembly fuel rod matrix. Consequently, solid rods fabricated from Zircaloy or Type 304 stainless steel must replace any fuel rod that is removed.

The shielding evaluations show that the Combustion Engineering Type A has the largest dose rates. The United Nuclear assemblies are evaluated for a source term based on an initial enrichment of 4.0 weight percent, a maximum burnup of 32,000 megawatt days per metric ton of uranium, and a minimum cool time of 13 years after reactor discharge. Exxon fuel with Zircaloy or stainless steel hardware is evaluated at 36,000 MWD/MTU and 9 or 16-years cool time, respectively, with an initial enrichment of 3.5 wt %. Westinghouse fuel is evaluated at 32,000 MWD/MTU and 21-years cool time with an initial minimum enrichment of 4.94 wt %. Combustion Engineering fuel is evaluated for a source term based on an initial minimum enrichment of 3.7 wt %, a maximum burnup of 36,000 megawatt days per metric ton of uranium, and a minimum cool time of 8.1 years after reactor discharge. For Combustion Engineering assemblies at a maximum burnup of 32,000 MWD/MTU and initial minimum enrichment of 3.5 wt %, a minimum cool time of 8 years is required.

The NAC-MPC maximum decay heat load is 12.5 kilowatts. This results in a maximum decay heat load for the design basis fuel assemblies of 0.347 kilowatt per assembly, based on 36 fuel assemblies per canister.

2.1.2 Reconfigured Fuel Assembly

One or more transportable storage canisters may hold Reconfigured Fuel Assemblies containing intact or damaged spent fuel rods and fuel debris. The Reconfigured Fuel Assembly may consist of up to 64 rod segments or whole rods. The rods, or rod segments, are held in individual tubes in an 8 by 8 array. The array of tubes is positioned in a stainless steel container having the same external dimensions as a standard fuel assembly. It has a top end fitting that has the same configuration as a standard fuel assembly. The container is closed on the top and bottom ends by perforated plates, which act as a barrier to the release of gross particles to the canister, but allow the draining and drying of the container. The tubes are stainless steel and are closed on each end by a plug. Each plug has a small hole drilled through it. The perforated plate screens the drilled hole. The hole allows the draining and drying of the individual tubes during routine closing of the canister. The perforated plate precludes the release of gross particles to the canister. The effects of the RFA container are evaluated in the appropriate sections. The structural, thermal, shielding, confinement, and criticality effects of the Reconfigured Fuel Assembly are bounded by those of an intact fuel assembly.

The physical parameters of the Reconfigured Fuel Assembly are provided in Table 2.1-2.

2.1.3 Stainless Steel-Clad Fuel

The short-term and long-term temperature limits for stainless steel-clad fuel are derived based on the limits presented in EPRI report TR-106440, "Evaluation of Expected Behavior of LWR Stainless Steel-Clad Fuel in Long-Term Dry Storage," April 1996. In this report, the potential failure modes in both wet and dry storage environments were assessed to develop the bounding conditions for the prevention of any potential cladding degradation phenomena and cladding failure modes of stainless steel-clad fuel. The potential cladding degradation mechanisms evaluated include: general corrosion, stress corrosion cracking, localized corrosion, mechanical failures and chemical/metallurgical-based failure mechanisms. The EPRI report is based on several types of stainless steel cladding, including 304, modified 348, 348 and 348H, which includes Yankee Class fuel.

The long-term temperature limit is conservatively established as 340°C for the NAC-MPC System for both stainless steel and Zircaloy-clad fuel. This value is significantly lower than the 430°C temperature limit defined by the EPRI report for stainless steel cladding. The short-term temperature limit for stainless steel-clad fuel is 430°C for the NAC-MPC System. This short-term limit is also conservatively used for Zircaloy-clad fuel.

Table 2.1-1 Yankee Class Fuel Parameters

	Combustion Engineering Type A	Combustion Engineering Type B	Exxon Type A	Exxon Type B	Exxon Type A	Exxon Type B	Westinghouse Type A	Westinghouse Type B	United Nuclear Type A	United Nuclear Type B
ASSEMBLY CONFIGURATION										
Assembly Length (cm)	283.9	283.9	283.3	283.3	283.9	283.9	282.6	282.6	282.4	282.4
Assembly Width (cm)	19.2	19.2	19.3	19.3	19.3	19.3	19.3	19.3	19.4	19.4
Assembly Cross Section (cm)	18.1	18.1	18.2	18.2	18.2	18.2	18.2	18.2	18.2	18.2
Assembly Array	16x16	16x16	16x16	16x16	16x16	16x16	18x18	18x18	16x16	16x16
Assembly Weight (kg)	352	350.6	372	372	372	372	408.2	408.2	385.5	385.5
Enrichment-wt. % ^{235}U Maximum	3.90	3.90	4.00	4.00	4.00	4.00	4.94	4.94	4.00	4.00
Minimum	3.70	3.70	3.50	3.50	3.50	3.50	4.94	4.94	4.00	4.00
Initial Fuel Weight ($KgUO_2$/Assembly)	264.8	264.1	268.3	266.6	266.2	265.0	311	310	273.8	272.6
Initial Heavy Metal (KgU/Assembly)	233.4	232.8	236.5	235	234.5	233.6	274.1	273.2	241.3	240.3
Max. Burnup (MWD/MTU)	36,000[1]	36,000[1]	36,000	36,000	36,000	36,000	32,000	32,000	32,000	32,000
Min. Cool Time (yr)	8.1[1]	8.1[1]	16.0	16.0	9.0	9.0	21.0	21.0	13.0	13.0
Max. Decay Heat (kW)	0.347[1]	0.347[1]	0.269	0.269	0.331	0.331	0.264	0.264	0.257	0.257
FUEL ROD CONFIGURATION										
Fuel Rod Pitch (cm)	1.2	1.2	1.2	1.2	1.2	1.2	1.1	1.1	1.2	1.2
Rod Length (cm)	242.3	242.3	242.2	242.2	242.2	242.2	237.7	237.7	242.1	242.1
Active Fuel Length (cm)	231.1	231.1	231.1	231.1	231.1	231.1	234.0	234.0	231.1	231.1
Rod OD (cm)	0.9	0.9	0.9	0.9	0.9	0.9	0.86	0.86	0.9	0.9
Clad ID (cm)	0.8	0.8	0.8	0.8	0.8	0.8	0.76	0.76	0.8	0.8
Pellet OD (cm)	0.79	0.79	0.79	0.79	0.79	0.79	0.75	0.75	0.79	0.79
Rods per Assembly	231	230	231	230	231	230	305	304	237	236
Fuel Material	UO2	UO2	UO2	UO2	UO2	UO2	UO2	UO2	UO2	UO2
Clad Material	Zircaloy	Zircaloy	Zircaloy	Zircaloy	Zircaloy	Zircaloy	SS 348	SS 348	Zircaloy	Zircaloy
Fill Gas	Helium	Helium	Helium	Helium	Helium	Helium	Air	Air	Helium	Helium
Fill Gas Pressure (psi)	315	315	125	125	125	125	0.0	0.0	140	140
DISPLACEMENT ROD CONFIGURATION										
Displacement Rod Material	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	Zircaloy - 4	Zircaloy - 4
Displacement Rod Diameter (cm)	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	0.9	0.9
Displacement Rod Length (cm)	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	242.1	242.1
Number Per Assembly	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	2	2

1. Combustion Engineering fuel may be loaded at a cool time of 8.0 years with a maximum burnup of 32,000 MWD/MTU and a minimum enrichment of 3.5 wt%. The maximum decay heat for this assembly is 0.304 kW.

Table 2.1-1 Yankee Class Fuel Parameters (Continued)

	Combustion Engineering	Combustion Engineering	Exxon	Exxon	Exxon	Exxon	Westinghouse	Westinghouse	United Nuclear	United Nuclear
	Type A	Type B	Type A	Type B	Type A	Type B	Type A	Type B	Type A	Type B
GUIDE BAR CONFIGURATION										
Guide Bar Material	Zircaloy - 4	Zircaloy - 4	SS 304L	SS 304L	Zircaloy	Zircaloy	--	--	N/A[2]	N/A[2]
Guide Bar Width (cm)	1.1	1.1	1.1	1.1	1.1	1.1	--	--	N/A[2]	N/A[2]
Guide Bar Length (cm)	245.2	245.2	244.6	244.6	244.6	244.6	--	--	N/A[2]	N/A[2]
Assembly Configuration	Type A	Type B	Type A	Type B	Type A	Type B	Type A	Type B	Type A	Type B
Top Nozzle Material	SS 304	SS 304	SS 304	SS 304	SS 304	SS 304	SS 304	SS 304	SS 304	SS 304
Bottom Nozzle Material	SS 304	SS 304	SS 304	SS 304	SS 304	SS 304	SS 304	SS 304	SS 304	SS 304
Upper Plenum Spring Material	SS 302	SS 302	Inconel X 750	Inconel X 750	Inconel X 750	Inconel X 750	N/A	N/A	Inconel X 750	Inconel X 750
Lower Plenum Spacer Material	N/A	N/A	SS 304	SS 304	SS 304	SS 304	N/A	N/A	SS 304	SS 304
Shroud Material	--	--	--	--	--	--	--	--	SS 304	SS 304
Top Nozzle Length (cm)	20.0	20.0	19.8	19.8	20.4	20.4	22.2	22.2	18.9	18.9
Bottom Nozzle Length (cm)	18.3	18.3	18.9	18.9	18.9	18.9	22.2	22.2	18.9	18.9
Upper Plenum Length (cm)	4.9	4.9	4.9	4.9	4.9	4.9	4.6	4.6	4.8	4.8
Lower Plenum Length (cm)	N/A	N/A	3.2	3.2	3.2	3.2	N/A	N/A	3.1	3.1
Shroud Length (cm)	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	246.9	246.9
Shroud Thickness (cm)	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	0.09	0.09
Top Nozzle Weight (kg)	5.50	5.50	6.70	6.70	6.70	6.70	6.70	6.70	17.02[4]	17.02[4]
Bottom Nozzle Weight (kg)	9.10	9.10	5.18	5.18	5.18	5.18	5.20	5.20	13.21[4]	13.21[4]
Upper Plenum Spring Weight (g)	3.3	3.3	3.3	3.3	3.3	3.3	N/A	N/A	3	3
Lower Plenum Spring Weight (g)	N/A	N/A	4.5	4.5	4.5	4.5	N/A	N/A	7.5	7.5
Grid Spacer Material	Zirc-4/Inconel 625[3]	Zirc-4/Inconel 625[3]	SS 304L	SS 304L	Zircaloy -4	Zircaloy -4	N/A	N/A	Inconel 718	Inconel 718
Grid Spacer Weight (g)	590/960	590/960	622.3	622.3	634.3	648.4	N/A	N/A	0	0
Number of Grid Spacers	6	6	6	6	6	6	N/A	N/A	6	6

2. United Nuclear assemblies fabricated from Zircaloy incorporate displacement rods 95.32-inches long and 0.365-inch in outside diameter.
3. Five grid spacers are Zircaloy 4. The bottom spacer is Inconel 625.
4. Estimated weight.

Table 2.1-2 Yankee Class Reconfigured Fuel Assembly Parameters

Parameter	CE Type A/B	Exxon Type A/B	Exxon Type A/B	West. Type A/B	United Nuclear Type A/B
ASSEMBLY CONFIGURATION					
Assembly Array	8x8	8x8	8x8	8x8	8x8
Max. Enrichment (wt % ^{235}U)	3.90	4.00	3.70	4.94	4.00
Max. kgU*	66.33	66.33	66.33	60.21	66.33
FUEL ROD CONFIGURATION (EACH ROD PLACED WITHIN ENCAPSULATING ROD)					
Rod Pitch (cm)	1.905	1.905	1.905	1.905	1.905
Active Fuel Length (cm)	231.1400	231.1400	231.1400	233.9975	231.1400
Rod OD (cm)	0.9271	0.9271	0.9271	0.8636	0.9271
Clad ID (cm)	0.8052	0.8052	0.8052	0.7569	0.8052
Pellet OD (cm)	0.7887	0.7887	0.7887	0.7468	0.7887
Diametrical Gap (cm)	0.0165	0.0165	0.0165	0.0102	0.0165
Max Rods per Assembly	64	64	64	64	64
Fuel Material	UO_2	UO_2	UO_2	UO_2	UO_2
Clad Material	Zircaloy	Zircaloy	Zircaloy	SS 348	Zircaloy
ENCAPSULATING ROD					
Rod OD (cm)	1.27	1.27	1.27	1.27	1.27
Rod ID (cm)	1.1278	1.1278	1.1278	1.1278	1.1278
Rod Material	SS-304	SS-304	SS-304	SS-304	SS-304

* Maximum kgU based on 95% of UO_2 theoretical density for the fuel pellet stack density.

Note: Intact or broken fuel rods may be contained in the reconfigured fuel assembly.

2.2 Design Criteria for Environmental Conditions and Natural Phenomena

The design criteria defined in this section identifies the site environmental conditions and natural phenomena to which the storage system could reasonably be exposed during the period of storage. Analyses to demonstrate that the NAC-MPC design meets the design criteria defined in this chapter are presented in later chapters of this report.

2.2.1 Tornado and Wind Loadings

The NAC-MPC may be stored on an unsheltered reinforced concrete storage pad at an ISFSI site. This storage configuration exposes the NAC-MPC to tornado and wind loading.

2.2.1.1 Applicable Design Parameters

The design basis tornado and wind loading is defined based on Regulatory Guide 1.76 Region 1 and NUREG-0800. The tornado and wind loading criteria are presented in Table 2.2-1.

2.2.1.2 Determination of Forces on Structures

Tornado wind forces on the NAC-MPC are calculated by multiplying the dynamic wind pressure by the frontal area of the cask normal to the wind direction. Wind forces are applied to the cask in the wind direction. No streamlining is assumed. The evaluation of wind loading and tornado missile effects on the NAC-MPC is presented in Section 11.2.13.

2.2.1.3 Tornado Missiles

The design basis tornado missile impacts are defined in Paragraph 4, Subsection III, Section 3.5.1.4 of NUREG-0800. The design basis tornado is considered to generate three types of missiles that impact the cask at normal incidence:

1. Massive Missile - (Deformable w/high kinetic energy)	Weight = 1800 kg (3960 pounds) Frontal Area = 20 sq.-ft.
2. Penetration Missile - (Rigid hardened steel)	Weight = 125 kg (275 pounds) Diameter = 8.0 inches
3. Protective Barrier Missile - (Solid steel sphere)	Weight = 0.068 kg (0.15 pounds) Diameter = 1.0 inch

Each missile is assumed to impact the cask at a velocity of 126 miles per hour, horizontal to the ground, which is 35 percent of the maximum wind speed of 360 miles per hour. For missile impacts in the vertical direction, the assumed missile velocity is: (0.7)(126) = 88.2 miles per hour.

The detailed analysis of the NAC-MPC for the missile impacts applies the laws of conservation of momentum and conservation of energy to estimate the impact force on the cask as a function of the time of impact and the amount of missile deformation. Each missile impact is evaluated, and all missiles are assumed to impact in a manner that produces the maximum damage to the NAC-MPC.

2.2.2 Water Level (Flood) Design

The NAC-MPC may be exposed to a flood during storage on an unsheltered concrete storage pad at an ISFSI site. The source and magnitude of the probable maximum flood depend on several variables.

2.2.2.1 Flood Elevations

The NAC-MPC is evaluated for a maximum flood water depth of 50 feet above the base of the storage cask. The flood water velocity is considered to be 15 feet per second. Under these conditions, the NAC-MPC will not float, tip or significantly slide on the storage pad, and the confinement function will be maintained.

2.2.2.2 Phenomena Considered in Design Load Calculations

The occurrence of flooding at an ISFSI site is dependent upon the specific site location and the surrounding geographical features, natural and man-made. Flooding of an ISFSI site is highly improbable because of the extensive environmental impact studies that are performed during the selection of a site for a nuclear facility. Some possible sources of a flood at an ISFSI site are: (1) overflow from a river or stream, due to unusually heavy rain, snow-melt runoff, a dam or major water supply line break caused by a seismic event (earthquake); (2) high tides produced by a hurricane; and (3) a tsunami (tidal wave) caused by an underwater earthquake or volcanic eruption.

2.2.2.3 Flood Force Application

The evaluation of the NAC-MPC for a flood condition determines a maximum allowable flood water current velocity and a maximum allowable flood water depth. The criteria employed in the determination of the maximum allowable values are that cask sliding or tip-over will not occur, and that the canister material yield strength is not exceeded. The evaluation of the effects of flood conditions on the NAC-MPC is presented in Section 11.2.6.

The force of the flood water current on the NAC-MPC is calculated as a function of the current velocity by multiplying the dynamic water pressure by the frontal area of the cask that is normal to the current direction. The dynamic water pressure is calculated using Bernoulli's equation relating fluid velocity and pressure. The force of the flood water current is limited such that the overturning moment on the cask will be less than that required to tip the cask over.

2.2.2.4 Flood Protection

The inherent strength of the reinforced concrete cask component of the NAC-MPC provides a substantial margin of safety against any permanent deformation of the cask for a credible flood event at an ISFSI site. Therefore, no special flood protection measures for the NAC-MPC are necessary. The evaluation presented in Section 11.2.6 shows that for the design basis flood, the allowable stresses in the canister are not exceeded.

2.2.3 Seismic Design

The NAC-MPC may be exposed to a seismic event (earthquake) during storage on an unsheltered concrete pad at an ISFSI site. The seismic response spectra experienced by the cask will depend upon the geographical location of the specific site and the distance from the epicenter of the earthquake. The only significant effect of a seismic event on the NAC-MPC would be a possible tip-over; however, tip-over does not occur in the evaluated design basis earthquake. Seismic response of the NAC-MPC is presented in Section 11.2.2.

2.2.3.1 Input Criteria

The magnitude of the maximum seismic accelerations to which the NAC-MPC may be subjected to are site specific. 10 CFR 72.102 defines a 0.10 g horizontal ground motion design earthquake as the minimum allowable seismic design criteria, and 0.25 g is suggested for sites east of the Rocky Mountain front. The NAC-MPC is designed to 0.25 g horizontal and 0.167 g vertical seismic acceleration. This acceleration provides seismic qualification for a predominant number of nuclear facilities within the United States.

2.2.3.2 Seismic - System Analyses

The seismic ground acceleration that will cause the NAC-MPC to tip over is calculated in Section 11.2.2 using quasi-static analysis methods. Both horizontal and vertical acceleration components are considered in the analyses. These components are calculated and combined according to Section 3.7.1 of NUREG-0800. Evaluation of the consequences of a tip over event is provided in Section 11.2.12.

2.2.4 Snow and Ice Loadings

The criterion for determining design snow loads is based on ANSI/ASCE 7-93, Section 7.0. Flat roof snow loads apply and are calculated from the following formula:

$$p_f = 0.7 C_e C_t I p_g$$

where:

p_f = flat roof snow load (psf)
C_e = Exposure factor = 1.0
C_t = Thermal factor = 1.2
I = Importance factor = 1.2
p_g = ground snow load, (psf) = 100

The numerical values of C_e, C_t, I and p_g are obtained from Tables 18, 19, 20 and Figure 7, respectively, of ANSI/ASCE 7-93.

The exposure factor accounts for wind effects. The NAC-MPC is assumed to have a site location typical for siting Category C, which is defined to be "locations in which snow removal by wind cannot be relied on to reduce roof loads because of terrain, higher structures, or several trees nearby."

The thermal factor accounts for the importance of buildings and structures in relation to public health and safety. The NAC-MPC is conservatively classified as Category III.

Ground snow loads for the contiguous United States are given in Figures 5, 6 and 7 of ANSI/ASCE 7-93. A worst case value of 100 pounds per square foot (psf) was assumed.

Based on the above, the design criterion for snow and ice loads is:

$$\text{Flat Roof Snow Load, } p_f = (0.7)\,(1.0)\,(1.2)\,(1.2)\,(100) = 100.8 \text{ psf}$$

This load is bounded by the weight of the loaded transfer cask. The snow load is considered in the load combinations described in Section 3.4.4.2.2.

2.2.5 Combined Load Criteria

Each normal, off-normal and accident condition has a combination of load cases that defines the total combined loading for that condition. The individual load cases considered include thermal, seismic, external and internal pressure, missile impacts, drops, snow and ice loads, and/or flood water forces.

The load conditions to be evaluated for storage casks are identified in 10 CFR 72 and in the "Design Criteria for an Independent Spent Fuel Storage Installation (Dry Storage Type)" (ANSI/ANS 57.9 - 1992).

2.2.5.1 Load Combinations and Design Strength - Concrete Cask

The load combinations specified in ANSI/ANS 57.9 - 1992 for concrete structures are applied to the concrete casks as shown in Table 2.2-2. The live loads are considered to vary from 0 percent to 100 percent to ensure that the worst-case condition is evaluated. In each case, use of 100 percent of the live load produces the maximum load condition. The steel liner of the concrete cask is a stay-in-place form and it provides radiation shielding. The concrete cask is designed to the requirements of ACI 349.

2.2.5.2 Design Strength Reduction Factors - Concrete

In calculating the design strength of the NAC-MPC concrete body, nominal strength values are multiplied by a strength reduction factor in accordance with Section 9.3 of ACI 349.

2.2.5.3 Load Combinations and Design Strength - Canister and Basket

The canister is designed in accordance with the 1995 edition of the ASME Code, Section III, Subsection NB for Class 1 components. The basket structure is designed per ASME Code, Section III, Subsection NG, and structural buckling of the basket is evaluated per NUREG/CR–6322.

The load combinations for all normal, off-normal, and accident conditions and corresponding service levels are shown in Table 2.2-3. Levels A and D service limits are used for normal and accident conditions, respectively. Levels B and C service limits are used for off-normal conditions. The analysis methods allowed by the ASME Code are employed. Stress intensities caused by pressure, temperature, and mechanical loads are combined before comparing to ASME code allowables, which are listed in Table 2.2-4.

2.2.5.4 Design Strength - Transfer Cask

The transfer cask is a special lifting device and is designed and fabricated to the requirements of ANSI N14.6 and NUREG 0612 for the lifting trunnions and supports. The criteria are:

The combined shear stress or maximum tensile stress during the lift (with 10 percent dynamic load factor) shall be $\leq S_y/6$ and $S_u/10$ for a nonredundant load path, or shall be $\leq S_y/3$ and $S_u/5$ for redundant load paths.

The ferritic steel material used for the load bearing members of the transfer cask shall satisfy the material toughness requirements of ANSI N14.6, paragraph 4.2.6.

2.2.6 Environmental Temperatures

A temperature of 75°F was selected to bound all annual average temperatures in the United States, except the Florida Keys and Hawaii.

The 75°F normal temperature was used as the base for thermal evaluations. The evaluation of this environmental condition is discussed along with the thermal analysis models in Chapter 4.0. The thermal stress evaluation for the normal operating conditions is provided in Section 3.4.4. Normal temperature fluctuations are bounded by the severe ambient temperature cases that are evaluated as off-normal and accident conditions.

Off-normal, severe environmental conditions were defined as -40°F with no solar loads and 100°F with solar loads. An extreme environmental condition of 125°F with maximum solar loads is evaluated as an accident case to show compliance with the maximum heat load case required by ANSI-57.9 (Section 11.2.10). Thermal performance was also evaluated for the cases of: (1) half the air inlets blocked; and (2) all air inlets and outlets blocked. Thermal analyses for these cases are presented in Sections 11.1.1 and 11.2.8. The evaluation based on ambient temperature conditions is presented in Section 4.4.

The design basis temperatures used in the NAC-MPC analysis are shown below. Solar insolance is as specified in 10 CFR 71.71 and Regulatory Guide 7.8.

Condition	Ambient Temperature	Solar Insolance
Normal	75°F	yes
Off-Normal - Severe Heat	100°F	yes
Off-Normal - Severe Cold	-40°F	no
Accident - Extreme Heat	125°F	yes

Table 2.2-1 Tornado and Wind Loading Criteria

Environmental Condition	Limit
Rotational Wind Speed, mph	290
Translational Wind Speed, mph	70
Maximum Wind Speed, mph	360
Radius of Max. Wind Speed, ft.	150
Pressure Drop, psi	3.0
Rate of Pressure Drop, psi/sec	2.0

Table 2.2-2 Load Combinations for the NAC-MPC Vertical Concrete Cask

Load Combination	Condition	Dead	Live	Wind	Thermal	Seismic	Tornado/ Missile	Drop/ Impact	Flood
1	Normal	1.4D	1.7L						
2	Normal	1.05D	1.275L		$1.275T_o$				
3	Normal	1.05D	1.275L	1.275W	$1.275T_o$				
4	Off-Normal & Accident	D	L		T_a				
5	Accident	D	L		T_o	E_{ss}			
6	Accident	D	L		T_o			A	
7	Accident	D	L		T_o				F
8	Accident	D	L		T_o		W_t		

Load Combinations are from ANSI 57.9 and ACI 349.

D = Dead Load
L = Live Load
W = Wind
T_o = Normal Temperature
F = Flood

T_a = Off- Normal or Accident Temperature
E_{ss} = Design Basis Earthquake
W_t = Tornado/Tornado Missile
A = Drop/Impact

Table 2.2-3 Load Combinations for the Transportable Storage Canister

LOAD		NORMAL			OFF-NORMAL					ACCIDENT					
ASME Service Level		A			B		C			D					
Load Combinations		1	2	3	1	2	3	4	5	1	2	3	4	5	6
DEAD WEIGHT	Canister with fuel	X	X	X	X	X	X	X	X	X	X	X	X	X	X
THERMAL	In Storage Cask 75° F Ambient	X		X				X		X	X	X	X	X	
	In Transfer Cask 75° F Ambient		X		X		X								X
	In Storage Cask -40°F or 100°F Ambient					X			X						
INTERNAL PRESSURE	Normal	X	X	X				X		X	X	X	X		
	Off-Normal				X	X	X		X						
	Accident													X	X
HANDLING LOAD	Normal		X	X	X										
	Off-Normal						X	X	X						
DROP/IMPACT	Accident									X					
SEISMIC	Accident										X				
FLOOD	Accident											X			
TORNADO	Accident												X		

Table 2.2-4 Structural Design Criteria for Components Used in the Transportable Storage Canister

	Component	Criteria
1.	Normal Operations: Service Level A Canister: ASME Section III, Subsection NB Basket: ASME Section III, Subsection NG Canister Lifting Devices ANSI N14.6 and NUREG 0612	$P_m \leq S_m$ $P_L + P_b \leq 1.5\ S_m$ $P_L + P_b + Q \leq 3S_m$ Redundant load path: combined shear or max. tensile stress $\leq S_u/5$ and $S_y/3$
2.	Off-Normal Operations: Service Level B Canister: ASME Section III, Subsection NB	$P_m < 1.1\ S_m$ $P_L + P_b < 1.65\ S_m$
3.	Off-Normal Operations: Service Level C Canister: ASME Section III, Subsection NB Basket: ASME Section III, Subsection NG	Subsection NB Allowables: $P_m < 1.2\ S_m$ or S_y (whichever is greater) $P_L + P_b < 1.8\ S_m$ or $1.5\ S_y$ (whichever is less) Note: Level C allowables for Subsection NG are larger than those for Level C per Subsection NB. Therefore, it is conservative to employ Subsection NB allowables for the basket.
4.	Accident Conditions, Service Level D Canister: ASME Section III, Subsection NB Basket: ASME Section III, Subsection NG	$P_m \leq 2.4\ S_m$ or $0.7\ S_u$ (whichever is less) $P_L + P_b \leq 3.6\ S_m$ or $1.05\ S_u$ (whichever is less)
5.	Basket Structural Buckling	NUREG/CR-6322

2.3 Safety Protection Systems

The NAC-MPC relies upon passive systems to ensure the protection of public health and safety, except in the case of fire or explosion. As discussed in Section 2.3.6, fire and explosion events are effectively precluded by site administrative controls that prevent the introduction of flammable and explosive materials. The use of passive systems provides protection from mechanical or equipment failure.

2.3.1 General

The NAC-MPC is designed for safe, long-term storage of spent nuclear fuel. The NAC-MPC will survive all of the evaluated normal, off-normal, and postulated accident conditions without release of radioactive material or excessive radiation exposure to workers or the general public. The major design considerations that have been incorporated in the NAC-MPC system to assure safe long-term fuel storage are:

1. Continued confinement in postulated accidents.
2. Thick concrete and steel biological shield.
3. Passive systems that ensure reliability.
4. Inert atmosphere to provide corrosion protection for stored fuel cladding.

Each NAC-MPC system storage component is classified with respect to its function and corresponding effect on public safety. In accordance with Regulatory Guide 7.10, each system component is assigned safety classification into Category A, B or C, as shown in Table 2.3-1. The safety classification is based on review of each component's function and the assessment of the consequences of component failure following the guidelines of NUREG/CR-6407, "Classification of Transportation Packaging and Dry Spent Fuel Storage System Components According to Importance to Safety."

Category A - Components critical to safe operations whose failure or malfunction could directly result in conditions adverse to safe operations, integrity of spent fuel or public health and safety.

Category B - Components with major impact on safe operations whose failure or malfunction could indirectly result in conditions adverse to safe operations, integrity of spent fuel or public health and safety.

Category C - Components whose failure would not significantly reduce the packaging effectiveness and would not likely result in conditions adverse to safe operations, integrity of spent fuel, or public health and safety.

As discussed in the following sections, the NAC-MPC design incorporates features addressing the above design considerations to assure safe operation during fuel loading, handling, and storage.

2.3.2 Protection by Multiple Confinement Barriers and Systems

2.3.2.1 Confinement Barriers and Systems

The radioactivity that the NAC-MPC must confine originates from the spent fuel assemblies to be stored and residual contamination that may remain inside the canister as a result of contact with the water in the fuel pool where the canister loading is conducted.

The NAC-MPC is designed to confine the radioactive fuel. The canister is closed by welding. The shield lid weld is pressure tested. All of the field-installed welds are liquid penetrant examined following the root and final weld passes. The shield lid weld is leak tested to 1.0×10^{-7} cubic centimeters per second (air). The closure of the canister structural lid, which provides a redundant closure over the shield lid and port covers, is accomplished by multi-pass welding that is tested by liquid penetrant examination on the root, intermediate and final passes. The longitudinal and girth welds and bottom welds of the canister shell are full penetration welds that are radiographed or ultrasonically inspected during fabrication.

The canister welds are an impenetrable boundary to the release of fission gas products during the period of storage. There are no evaluated normal, off-normal, or accident conditions that result in the breach of the canister and the subsequent release of fission products. The canister is designed to withstand a postulated drop accident in a transportation cask without precluding the subsequent removal of the fuel (i.e., the fuel tubes do not deform such that they bind the fuel).

Personnel radiation exposure during handling and closure of the canister is minimized by the following steps:

1. Placing the shield lid on the canister while the transfer cask and canister are under water in the fuel pool.
2. Decontaminating the exterior of the transfer cask prior to draining the canister to preserve the shielding benefit of the water.
3. Using temporary shielding.
4. Using a retaining ring on the transfer cask to ensure that the canister is not raised out of the shield provided by the transfer cask.
5. Placing a shielding ring over the annular gap between the transfer cask and the canister.

2.3.3 Protection by Equipment and Instrumentation Selection

The NAC-MPC is a passive storage system that does not rely on equipment or instruments to preserve public health or safety and to meet its safety functions in long-term storage. The system employs support equipment and instrumentation to facilitate operations. These items and the actions taken to assure performance are described below.

2.3.3.1 Equipment

The only important-to-safety equipment employed in the use and operation of the NAC-MPC is the lifting yoke used to lift the transfer cask. The transfer cask lifting yoke is designed to meet the requirements of ANSI N14.6 and NUREG-0612. It is single failure-proof by design. The lifting yoke is proof load tested to 300 percent of design load when fabricated. The lifting yoke is inspected for visible defects prior to each use and is inspected annually.

Additional handling equipment (such as trailers, skids, air pads, portable cranes, or cask transporters) are not important to safety as the NAC-MPC system is designed to withstand the failure of any of these components.

2.3.3.2 Instrumentation

A remote temperature measuring system is employed to measure the outlet air temperature of the NAC-MPC in long-term storage. The outlet temperature is recorded daily as a check of the thermal performance of the heat rejection capability of the storage cask. The outlet temperature is expected to increase in the unlikely event that one or more inlet or outlet ventilation ports become blocked.

The inlet and outlet ports are visually inspected each day during the same walk-through in which the temperatures are recorded. This visual inspection assists in ensuring that the temperature measuring system is continuing to perform as expected.

The canister shield lid weld is helium leak tested during closure. The leak detector is checked against a known helium source immediately prior to, and after, use to preclude unknown leak detector failure.

2.3.4 Nuclear Criticality Safety

The primary nuclear criticality safety design criterion of the NAC-MPC is to provide features that ensure that the cask remains subcritical under normal, off-normal, and accident conditions. Neutron poison sheets (BORAL) are employed in the basket design to capture thermalized neutrons, and preclude uncontrolled fission events. BORAL sheets are attached to each side of each fuel tube. These sheets are mechanically supported by the fuel tube structure to ensure that the poison sheets remain in place during the design basis normal, off-normal, and accident events.

The efficiency of the BORAL sheets in preserving nuclear criticality safety is demonstrated by the Criticality Evaluation presented in Chapter 6.

2.3.4.1 Error Contingency Criterion

The design of the canister and fuel basket is such that, under all conditions, the highest neutron multiplication factor (k_{eff}) will be less than 0.95. The criticality evaluation for the design basis fuel is presented in Section 6.4. Assumptions made in the analyses used to demonstrate conformance to this criterion include:

1. Most reactive Yankee Class fuel assembly type with maximum ^{235}U loading;
2. 75 percent of the nominal ^{10}B loading in the BORAL;
3. Infinite array of casks in the X-Y (horizontal) plane;
4. Infinite fuel length with no inclusion of end leakage effects;
5. No structural material present in the assembly;
6. No credit taken for boron in the cask cavity or surrounding loading or storage area; and
7. No credit taken for fuel burnup or for the buildup of fission product neutron poisons.

These assumptions demonstrate adequate controls to assure subcriticality in the use of the NAC-MPC system.

2.3.5 Radiological Protection

The NAC-MPC system, in keeping with the As Low As Reasonably Achievable (ALARA) philosophy, is designed to minimize, to the extent practicable, operator radiological exposure.

2.3.5.1 Access Control

Access to an NAC-MPC ISFSI site is controlled by a peripheral fence to meet the requirements of 10 CFR 72 and 10 CFR 20. Access to the storage area, and its designation as to the level of radiation protection required, is established by site procedure. The storage area will be surrounded by a fence, having lockable truck and personnel access gates. The fence will have intrusion-detection features as determined by the site procedure.

2.3.5.2 Shielding

The NAC-MPC is designed to provide an external side surface dose (gamma and neutron) of less than 50 mrem/hr (average) on the storage cask sides, 35 mrem/hr (average) top, and 100 mrem/hr at the air vent inlets. The transfer cask side wall contact dose rate limit is 200 mrem/hr (average). The design maximum dose rate at the top of the canister structural lid, with supplemental shielding, is 200 mrem/hr (average) to limit personnel exposure during canister closure operations.

Sections 72.104 and 72.106 of 10 CFR 72 set whole body dose limits for an individual located beyond the controlled area at 25 millirems per year (whole body) during normal operations and 5 rems (5,000 millirems) from any design basis accident. The analyses showing the actual NAC-MPC doses are included in Sections 5.0 and 11.0.

2.3.5.3 Ventilation Off-Gas

The NAC-MPC is passively cooled by radiant and natural convection heat transfer at the outer surface of the canister and natural convective heat transfer in the canister-concrete cask annulus. The bottom of the cask is conservatively assumed to be an adiabatic surface. The design criterion for the air-flow in the annulus is that the pressure difference, due to the buoyancy effect created by the heating of the air, is equal to the flow pressure drop. The details of the passive ventilation system design are provided in Section 4.0.

There are no radioactive releases during normal operations. Also, there are no credible accidents that cause significant releases of radioactivity from the NAC-MPC and, hence, there are no off-gas system requirements for the NAC-MPC during normal storage operation. The only time an off-gas system is required is during the canister drying phase. During this operation, the reactor off-gas system or a HEPA filter system will be used.

The surface of the canister is exposed to cooling air when the canister is placed in the storage cask. If the surface is contaminated, the possibility exists that contamination could be carried aloft by the cooling air stream. To ensure that the canister surface is free of contamination, pool water is prevented from contacting the canister exterior by filling the transfer cask/canister annular gap with clean water as the transfer cask is being lowered into the fuel pool.

Clean water is injected into the gap during the entire time the transfer cask is submerged. These steps preclude the intrusion of contaminated water into the canister annular gap.

Once the transfer cask is removed from the pool, a smear survey is taken of the exterior surface of the canister near the top. While no contamination is expected to be found, it is possible that the surface could be contaminated. The allowable upper limit on surface contamination is calculated in Section 12.2.1.4. If this limit is exceeded, then steps to decontaminate the canister surface must be taken and continued until the contamination is less than the allowable limit.

To facilitate decontamination, the canister is fabricated so that its exterior surface is smooth. There are no corners or pockets that could trap and hold contamination.

2.3.5.4 Radiological Alarm Systems

There are no radiological alarms required on the NAC-MPC. Justification for this is provided in analysis in Sections 5.0 (Shielding), 10.0 (Radiological Protection), and 11.0 (Accident Analysis).

Typically, total radiation exposure due to the ISFSI installation is determined by the use of Thermo-Luminescent Detectors (TLDs) mounted at convenient locations on the ISFSI fence. The TLDs are read quarterly to provide a record of boundary dose.

2.3.6 Fire and Explosion Protection

Fire and explosion protection of the NAC-MPC is primarily provided by administrative controls applied at the site, which preclude the introduction of any explosive and any excessive flammable materials into the ISFSI area.

2.3.6.1 Fire Protection

A major ISFSI fire is not considered credible, since there is very little material near the casks that could contribute to a fire. The concrete cask is largely impervious to incidental thermal events. Administrative controls will be put in place to ensure that the presence of combustibles is minimized. A hypothetical fire event is evaluated as an accident condition in Section 11.2.5.

2.3.6.2 Explosion Protection

The cask and associated systems are analyzed to ensure their proper function under an overpressure condition. As described in Section 11.2.3, in the evaluated 22 psig over pressure condition, stresses in the canister remain below allowable limits and there is no loss of confinement. These results are conservative as the canister is protected from direct over-pressure conditions by the concrete storage cask.

For the same reasons as the fire condition, a severe explosion on an ISFSI site is not considered credible. The evaluated over-pressure is considered to bound any explosive over pressure resulting from an industrial explosion at the boundary of the owner-controlled area.

Table 2.3-1 Safety Classification of NAC-MPC Components

Drawing Number	Sh	Title	Rev	Item No.	Component	Function	Safety Class
455-861	1	Weldment, Structure	3	26	Baffle	Heat Transfer	B
				-	Outlet (4)	Heat Transfer	B
				20	Shield Plate	Shielding	B
				17	Nelson Stud	Structural	C
				16	Base Plate	Structural	B
				15	Stand	Structural	B
				-	Inlet (4)	Heat Transfer	B
				12	Bottom	Structural	B
				11	Shield Ring	Shielding	B
				10	Cover	Operations	B
				-	Jack (Leveling)	Operations	C
				3	Support Ring	Structural	B
				2	Top Flange	Structural	B
				1	Shell	Structural	B
455-862	1	Loaded Vertical Concrete Cask	2	9	Cover	Operations	B
				8	Insulation	Operations	B
				7	Washer (Lid Bolt)	Operations	C
				6	Lid Bolt	Operations	B
				5	Lid	Operations	B
				4	Shield Plug	Shielding	B
455-866	1	Reinforcing Bar and Concrete Placement	0	18	Name Plate	Operations	C
				17	Outlet Screen	Operations	C
				16	Inlet Screen	Operations	C
				15	Concrete	Shielding/ Structural	B
				13	Liner Weldment	Shielding/ Structural	B

Table 2.3-1 Safety Classification of NAC-MPC Components (Continued)

Drawing Number	Sh	Title	Rev	Item No.	Component	Function	Safety Class
455-866 (continued)	1	Reinforcing Bar and Concrete Placement	0	-	Reinforcing Bar	Structural	B
455-870	1	Canister, Shell	3	3	Location Nut	Operations	B
				2	Bottom	Structural/ Confinement	A
				1	Shell	Structural/ Confinement	A
455-871	1	Details, Canister	2	8	Key	Operations	C
				7	Port Cover (Two)	Confinement/ Operations	B
				6	Valved Nipple (Two)	Operations	C
				5	Structural Lid	Structural	A
				3	Shield Lid	Shielding	B
				2	Backing Ring	Structural	C
				1	Lid Support Ring	Structural	B
455-872	1	Assembly, Transportable Storage Canister	5	3	Drain Tube Assembly	Operations	C
455-881	1	PWR Fuel Tube	2	4	Flange	Structural	A
				3	Cladding	Criticality Control	A
				2	BORAL	Criticality Control	A
				1	Tubing	Structural/ Criticality	A
455-887		Drawing Deleted					
455-888		Drawing Deleted					
455-891	1	Bottom Weldment, Fuel Basket	0	-	Support	Structural	B
				2	Circular Pad	Structural	C
				1	Plate	Structural	A

Table 2.3-1 Safety Classification of NAC-MPC Components (Continued)

Drawing Number	Sh	Title	Rev	Item No.	Component	Function	Safety Class
455-892	1	Top Weldment, Fuel Basket	1	4	Baffle	Structural	A
				3	Support	Structural	A
				2	Support Ring	Structural	B
				1	Plate	Structural	A
455-893	1	Support Disk and Misc. Basket Details	3	7	Top Spacer	Structural	B
				6	Split Spacer	Structural	B
				5	Tie Rod	Structural	A
				4	Top Nut	Structural	A
				3	Bottom Spacer	Structural	B
				2	Spacer	Structural	A
				1	Plate	Structural	A
455-894	1	Heat Transfer Disk	1	1	Heat Transfer Disk	Heat Transfer	A
455-895	1	Fuel Basket Assembly	2	13	Flat Washer	Structural	B
				4	Drain Tube Sleeve	Operations	C
455-860	1	Assembly, Transfer Cask	3	17	Connector	Operations	C
				16	Fill/ Drain Line	Operations	C
				15	Retaining Ring Bolt	Operations	B
				14	Retaining Ring	Operations	B
				13	Door Lock Bolt	Operations	C
				11-12	Shield Door	Structural/Shielding	B
				10	Door Rail	Operations	B
				9	Top Plate	Structural	B
				8	Neutron Shield	Shielding	B
				7	Scuff Plate	Operations	C
				6	Trunnion Cap	Operations	C
				5	Trunnion	Structural	B

Table 2.3-1 Safety Classification of NAC-MPC Components (Continued)

Drawing Number	Sh	Title	Rev	Item No.	Component	Function	Safety Class
455-860 (continued)	1	Assembly, Transfer Cask	3	4	Outer Shell	Structural	B
				3	Gamma Shield Brick	Shielding	B
				2	Inner Shell	Structural	B
				1	Bottom Plate	Structural	B
YR-00-060	1	Yankee – Class Reconfigured Fuel Assembly	1	1	Shell Weldment	Structural	A
				2	Top End Fitting Assembly	Structural/ Criticality	A
				3	Bottom End Fitting Assembly	Structural/ Criticality	A
				4	Basket Assembly	Structural/ Criticality	A
				5	Top Nozzle Bolts	Structural/ Criticality	A
				6	Alignment Pin	Operations	C

2.4 Decommissioning Considerations

The principal elements of the NAC-MPC storage system are the vertical concrete cask (storage cask) and the transportable storage canister (canister).

The storage cask provides biological shielding and physical protection for the contents of the canister during long-term storage. The storage cask is not expected to become surface contaminated during use, except through incidental contact with other contaminated surfaces. Incidental contact could occur at the interior surface (liner) of the storage cask, the top surface that supports the transfer cask during loading and unloading operations, and the floor of the storage cask that supports the canister. All of these surfaces are carbon steel, and it is anticipated that these surfaces could be decontaminated as necessary for decommissioning. A layer of insulation and stainless steel is placed on the floor of the storage cask in order to separate the stainless steel canister bottom from the carbon steel storage cask bottom plate. The insulation rests on the storage cask carbon steel pedestal. The insulation is covered by a sheet of stainless steel. Contamination of these surfaces is expected to be minimal, since the canister is isolated from spent fuel pool water during loading in the pool and the transfer cask is decontaminated prior to transfer of the canister to the storage cask. In the unlikely event that the insulation became contaminated, it is not reasonable to expect that it could be decontaminated. Consequently, the insulation would have to be disposed of as surface-contaminated material.

The concrete that provides biological shielding is not expected to become contaminated during the period of use, as it does not come into contact with other contaminated objects or surfaces.

Activation of the carbon steel liner, concrete, support plates, and reinforcing bar could occur due to neutron flux from the stored fuel. Since the neutron flux rate is low, only minimal activation of carbon steel in the storage cask is expected to occur. The activity concentrations from activation of storage cask components are listed in Table 2.4-1. Table 2.4-1 includes the radiological significant isotopes, together with a total concentration of all activated nuclides in the respective component. The total concentrations listed include activities of radionuclides, which do not have any substantial contribution to radiation dose and are not specifically identified by 10 CFR 61 waste classification. In particular the isotope contributing the majority of the carbon steel total curie activity is ^{55}Fe, which decays by electron capture and is not of radiological concern.

Decommissioning of the storage cask would involve the removal of the canister and the subsequent disassembly of the storage cask. It is expected that the concrete would be broken up, and steel components segmented to reduce volume. Any contaminated or activated items are expected to qualify for near-surface disposal as low specific activity material.

The transportable storage canister is designed and fabricated to be suitable for use as a waste package for permanent disposal in a deep Mined Geological Disposal System, in that it meets the requirements of the DOE MPC Design Procurement Specification. The canister is fabricated from materials having high long-term corrosion resistance, and the canister contains no paints or coatings that could adversely affect the permanent disposal of the canister. Consequently, decommissioning of the canister would occur only if the fuel contained in the canister had to be removed, or if current requirements for disposal were to change. Decommissioning would require that the closure welds at the canister structural lid, shield lid and shield lid port covers be cut, so that the spent fuel could be removed. Removal of the contents of the canister would require that the canister be returned to a spent fuel pool or dry unloading facility, such as a hot cell. Closure welds can be cut either manually or with automated equipment, with the procedure being essentially the reverse of that used to initially close the canister.

Following removal of the contents, the canister could have significant internal contamination due to the contents, and may contain "crud" or other residual material in the bottom of the canister. Some effort may be required to remove the surface contamination prior to disposal; however, in practice, it would not be absolutely necessary to decontaminate the canister internals. Any contaminated canister and internal components are expected to qualify for near-surface disposal as low specific activity waste without internal contamination, as the internal contamination would consist only of by-product materials. Should internal decontamination be necessary, the canister and basket surfaces are smooth, and the design precludes the presence of crud traps, thus facilitating any required decontamination. Since the neutron flux rate from the stored fuel is low, only minimal activation of the canister is expected to occur. The activity concentrations from activation of canister components are listed in Table 2.4-1.

The unloaded canister could also qualify as a strong, tight container for other waste. In this case, the canister could be filled, within weight limits, with other qualified waste, closed and transported whole and complete to a near-surface disposal site. Use of the canister for this purpose could reduce decommissioning costs by avoiding decontamination, segmenting and repackaging.

The storage pad, fence and supporting utility fixtures are not expected to require decontamination as a result of use of the NAC-MPC system. The design of the cask and canister precludes the release of contamination from the contents over the period of use of the system. Consequently, these items may be reused or disposed of as locally generated clean waste.

Table 2.4-1 Activity Concentration Summary for the NAC-MPC Vertical Concrete Cask and Transportable Storage Canister

	Activity Concentration[1] (Ci/m^3)				
	Vertical Concrete Cask			Transportable Storage Canister	
Isotope	Shell[2]	Shield Plug	Lid	Shell	Lids
^{54}Mn	4.2E-03	8.77E-04	1.16E-04	6.0E-05	1.52E-05
^{60}Co	2.7E-05	1.40E-05	3.05E-06	2.3E-04	8.44E-05
^{59}Ni	---	---	---	2.6E-07	----
^{63}Ni	---	---	---	3.0E-01	1.10E-01
Total	8.9E-02	4.22E-02	1.19E-02	3.1E-01	1.13E-01

1. Seven days after removal of spent fuel.
2. Includes liner and concrete.

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3.0 STRUCTURAL EVALUATION

This section describes the design and analyses of the principal structural components of the NAC-MPC System under normal operating conditions. It demonstrates that the NAC-MPC System meets the requirements to assure confinement of contents, criticality control, radiological shielding, and contents retrievability as required by 10CFR72 for the design basis operating conditions. Off-normal and accident conditions are evaluated in Chapter 11.

3.1 Structural Design

3.1.1 Discussion

The NAC-MPC System consists of three major components: 1) the vertical concrete cask (storage cask); 2) the transportable storage canister (canister); and 3) the transfer cask. These components are shown in Figure 3.1-1. The principal structural member of the vertical concrete cask is the reinforced concrete shell. The principal structural members of the canister are the shell, structural lid, bottom plate, the welds joining these components, and the basket assembly. The primary structural components of the transfer cask are its trunnions, inner and outer steel walls and the bottom doors and their support rails. All of the components are shown on the license drawings provided in Section 1.5.

The concrete cask is a reinforced concrete cylinder with an outside diameter of 128 inches and an overall height of 160 inches. The internal cavity of the concrete cask is formed by a 3.5-inch thick cylindrical carbon steel liner having an inside diameter of 79 inches. The liner is a stay-in-place form. Its thickness is primarily determined by shielding requirements, but is related to the need to establish a practical limit to the diameter of the concrete shell. The concrete is Type II Portland Cement, having a nominal density of 140 lbs/ft^3, and a nominal compressive strength of 4000 psi. The inner and outer reinforcing bar assemblies are formed by vertical hook bars and horizontal hoop bars. The air flow path is formed by channels at the bottom that provide the entrance for cooling air, the air inlet ducts that admit the air to the storage cask interior cavity (i.e., the annular gap between the canister outer surface and the concrete cask liner interior surface, and the air outlet ducts.). A 5-inch thick carbon steel shield plug, that encloses a 1-inch thick layer of NS-4-FR neutron shield material, is installed in the concrete cask cavity above the canister. The plug is supported by a support ring welded to the liner. A 1.5-inch thick carbon steel lid provides a cover to protect the canister from adverse environmental conditions and postulated tornado driven missiles. The shield plug and lid provide shielding to reduce the skyshine radiation. The lid is bolted in place.

The canister consists of a cylindrical shell assembly closed at its top end by an inner shield lid and an outer structural lid. The canister contains a basket assembly that holds the spent fuel. The canister shell is 122.5 inches long and is fabricated from 304L stainless steel plate. The canister shield lid is 5-inch thick Type 304 stainless steel, and the structural lid is 3.0-inch thick Type 304L stainless steel. Both lids are welded to the canister shell to close the canister. The shield lid is supported from below, prior to welding, by a support ring. The structural lid is supported, prior to welding, by the shield lid. The bottom of the canister is a 1-inch thick, Type 304L stainless steel plate that is welded to the canister shell.

The basket assembly is designed to hold up to 36 Yankee Class fuel assemblies. It incorporates 22 Type 17-4 PH stainless steel support disks and 14 Type 6061-T6 aluminum alloy heat transfer disks. The remaining components of the basket assembly are Type 304 stainless steel. These disks, together with the top and bottom weldments, are positioned by tie rods (with spacers and washers) that extend the length of the basket and clamp the components together. The support disks provide heat removal and support the fuel tubes that pass through the disks. The heat transfer disks provide the heat removal capability, but are not considered to be structural components. The fuel tubes have an inside square dimension of 7.8 inches and a composite wall thickness of 0.14 inches. All walls of each fuel tube contain a sheet of BORAL neutron poison material. No structural credit is taken for the BORAL sheet.

A transportable storage canister containing spent fuel may also contain one or more Reconfigured Fuel Assemblies. The Reconfigured Fuel Assembly is designed to contain Yankee Class spent fuel rods, or portions thereof, which are classified as failed, and to maintain the geometric positions of the rods. The assembly has a capacity of 64 full length spent fuel rods in an eight by eight array of tubes. As shown in Figure 1.2-5, the reconfigured fuel assembly consists of a shell (square tube with end fittings), a basket assembly, and 64 fuel tubes. All of the materials are stainless steel.

The Yankee Class Reconfigured Fuel Assembly is designed to contain failed fuel rods, in fuel tubes, during all storage and transport conditions. The Reconfigured Fuel Assembly is designed to the requirements of ASME Boiler and Pressure Vessel Code, Section III, Article NG-3000 and

NUREG/CR-6322, "Buckling Analysis of Spent Fuel Baskets" and using the additional guidance contained in ASME Section III, Article NF-3000 and in ASME Section III, Appendix F. The structural evaluation of the Reconfigured Fuel Assembly is presented in Section 11.4.

The external dimensions of the Reconfigured Fuel Assembly are the same as those of other Yankee Class fuel assemblies. The weight of a loaded reconfigured fuel assembly (approximately 550 pounds) is less than the weight of other Yankee Class fuel assemblies (approximately 850 pounds). The maximum temperature of the Reconfigured Fuel Assembly components is determined by the thermal analyses presented in Section 4.4.

The Reconfigured Fuel Assembly has been evaluated and is capable of withstanding, within code allowable limits (Service Level A/B), a postulated end impact resulting in a deceleration of 20 g. It is also, when located in a fuel slot in the transportable storage container, capable of withstanding, within code allowable limits (Service Level A/B), a postulated side impact resulting in a deceleration of 20 g. This analysis bounds the design conditions of the Reconfigured Fuel Assembly for normal conditions of storage.

The Reconfigured Fuel Assembly has also been evaluated for accident conditions and is capable of withstanding, within code allowable limits (Service Level D), a postulated end impact resulting in a deceleration of 57 g. It is also, when located in a fuel slot in the transportable storage container, capable of withstanding, within code allowable limits (Service Level D), a postulated side impact resulting in a deceleration of 55 g. This analysis bounds the design conditions of the Reconfigured Fuel Assembly for accident conditions of storage.

Therefore, the structural evaluations of the NAC-MPC System containing other Yankee Class fuel assemblies (Chapters 3.0 and 11.0) bound those of the NAC-MPC System containing one or more Yankee Class Reconfigured Fuel Assemblies.

The following components are evaluated in this chapter:

- Canister lifting devices
- Canister shell, bottom, and structural lid
- Canister shield lid support ring
- Basket assembly
- Transfer cask trunnions, shells, retaining ring, bottom doors, and support rails
- Vertical concrete cask body
- Concrete cask steel components (reinforcement, liner, lid, bottom plate, bottom, etc.)

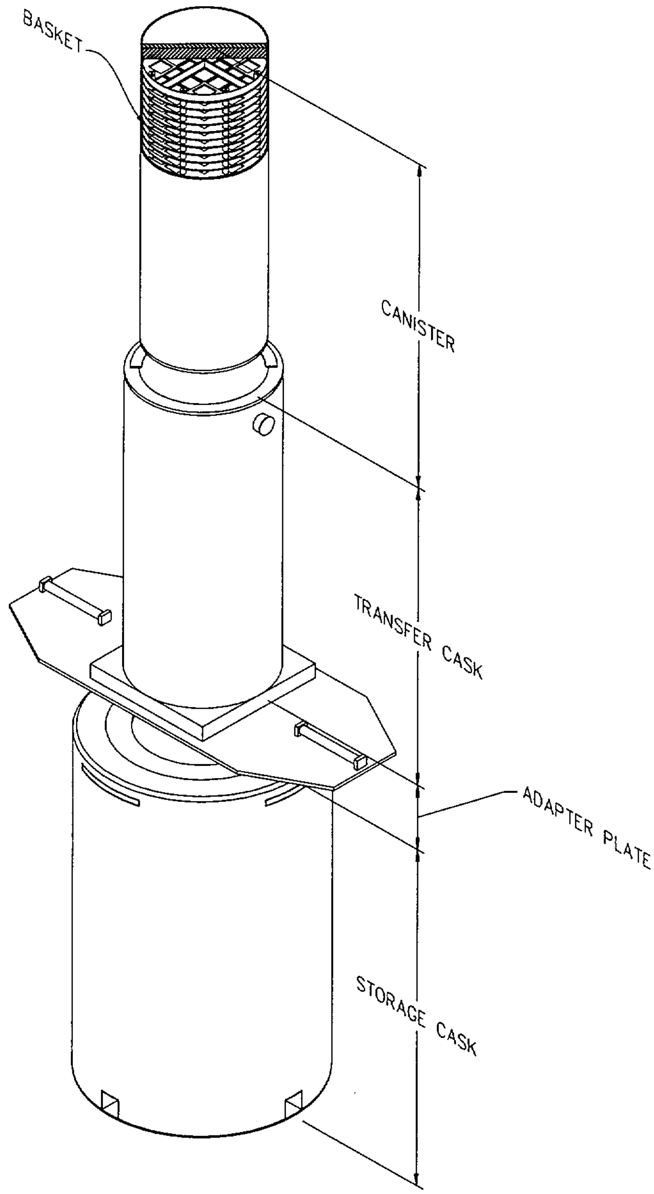
All other NAC-MPC system components shown on the drawings presented in Section 1.5 are either nonstructural or not classified as important to safety. They are appropriately included as loads in the evaluation of the components listed above.

The structural evaluations demonstrate that all of the NAC-MPC components meet their structural design criteria and are capable of safely storing the design basis spent fuel.

3.1.2 Design Criteria

The NAC-MPC structural design criteria is specified in Section 2.2. The load combinations of normal, off-normal, and accident loadings have been evaluated in accordance with ANSI 57.9 and ACI-349 for the concrete cask (see Table 2.2-2), and in accordance with the 1995 edition of the ASME Code, Section III, Division I, Subsection NB for Class 1 components for the canister (see Table 2.2-3). The basket is evaluated in accordance with ASME Code, Section III, Subsection NG, and NUREG-6322. The transfer cask and the lifting yoke are lifting devices that are designed to NUREG-0612 and ANSI N14.6.

Figure 3.1-1 Principal Components of the NAC-MPC System



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3.2 Weights and Centers of Gravity

The component weights and centers of gravity for the NAC-MPC system are summarized in Table 3.2-1.

Table 3.2-1 NAC-MPC System Weights and Centers of Gravity

Item Description	Calculated Weight (lbs)	Center of Gravity (inches above bottom of item)
Concrete Cask Lid	2,838	160.7
Concrete Cask Shield Plug	5,490	153.3
Canister Structural Lid	3,230	121.0
Canister Shield Lid	5,390	116.8
Transfer Adapter Plate	12,659	N/A
Canister (empty, without lid)	15,510	49.7
Canister (loaded with fuel with water and shield lid)	62,270	61.2
Canister (loaded with fuel with lid)	54,730	65.0
Concrete Cask (empty, with shield plug, and without lid)	148,526	80.5
Concrete Cask and Canister (loaded with fuel with lids)	206,094	83.2
Transfer Cask (empty)	80,743	57.0
Transfer Cask and Canister (empty, without lids)	96,253	58.0
Transfer Cask and Canister (loaded with fuel, with water and shield lid)	143,013	63
Transfer Cask and Canister (loaded with fuel, dry with lids)	135,473	64.1
Water in Canister (36 assemblies)	10,618	N/A
Fuel	30,600	55.7

3.3 Mechanical Properties of Materials

The mechanical properties of steels used in the fabrication of the NAC-MPC components are presented in Tables 3.3-1 through 3.3-8 and in Tables 3.3-12 and 3.3-13. The primary steels are Type 304 and Type 304L stainless steel, selected because of their high strength, ductility, resistance to corrosion and brittle fracture, and metallurgical stability for long-term storage. The mechanical properties for the 6061-T6 aluminum heat transfer disks in the fuel basket are provided in Table 3.3-9. The mechanical properties of the concrete are presented in Table 3.3-10. The mechanical properties of the neutron shield material, NS-4-FR, are presented in Table 3.3-11.

Table 3.3-1 Mechanical Properties of SA 240, A 479 Type 304 Stainless Steel

Property (units)	Temperature (°F)							
	-40	-20	+70	+200	+300	+400	+500	+750
Ultimate Strength[1] (ksi)	75.0	75.0	75.0	71.0	66.0	64.4	63.5	63.1
Yield Strength[2] (ksi)	30.0	30.0	30.0	25.0	22.5	20.7	19.4	17.3
Design Stress Intensity[3] (ksi)	20.0	20.0	20.0	20.0	20.0	18.7	17.5	15.6
Modulus of Elasticity[4] (ksi)	28.7E+3	28.7E+3	28.3E+3	27.6E+3	27.0E+3	26.5E+3	25.8E+3	24.4E+3
Alternating Stress[5] @ 10 cycles (ksi)	718.0	718.0	708.0	690.5	675.5	663.0	645.5	610.4
Alternating Stress[5] @ 10^6 cycles (ksi)	28.7	28.7	28.3	27.6	27.0	26.5	25.8	24.4
Coefficient of Thermal Expansion[6] (in/in/°F)	8.13E-6	8.19E-6	8.46E-6	8.79E-6	9.00E-6	9.19E-6	9.37E-6	9.76E-6
Poisson's Ratio[7]	0.275							
Density[8] (lbm/in^3)	0.288							

1 ASME Code, Section II, Part D, Table U.
2 ASME Code, Section II, Part D, Table Y-1.
3 ASME Code, Section II, Part D, Table 2A.
4 ASME Code, Section II, Part D, Table TM-1.
5 ASME Code, Section III, Appendix I, Table I-9.1.
6 ASME Code, Section II, Part D, Table TE-1.
7 Hanford, Volume 1, Design Data, Property Code 2110.
8 Hanford, Volume 1, Design Data, Property Code 3304.

Table 3.3-2 Mechanical Properties of SA 336, Type 304 Stainless Steel

Property (units)	Temperature (°F) -40	-20	+70	+200	+300	+400	+500	+750
Ultimate Strength[1] (ksi)	70.0	70.0	70.0	66.2	61.5	60.0	59.3	58.9
Yield Strength[2] (ksi)	30.0	30.0	30.0	25.0	22.5	20.7	19.4	17.3
Design Stress Intensity[3] (ksi)	20.0	20.0	20.0	20.0	20.0	18.7	17.5	15.6
Modulus of Elasticity[4] (ksi)	28.7E+3	28.7E+3	28.3E+3	27.6E+3	27.0E+3	26.5E+3	25.8E+3	24.4E+3
Alternating Stress[5] @ 10 cycles (ksi)	718.0	718.0	708.0	690.5	675.5	663.0	645.5	610.4
Alternating Stress[5] @ 10^6 cycles (ksi)	28.7	28.7	28.3	27.6	27.0	26.5	25.8	24.4
Coefficient of Thermal Expansion[6] (in/in/°F)	8.13E-6	8.19E-6	8.46E-6	8.79E-6	9.00E-6	9.19E-6	9.37E-6	9.76E-6
Poisson's Ratio[7]	0.275							
Density[8] (lbm/in^3)	0.288							

1 ASME Code, Section II, Part D, Table U.
2 ASME Code, Section II, Part D, Table Y-1.
3 ASME Code, Section II, Part D, Table 2A.
4 ASME Code, Section II, Part D, Table TM-1.
5 ASME Code, Section III, Appendix I, Table I-9.1.
6 ASME Code, Section II, Part D, Table TE-1.
7 Hanford, Volume 1, Design Data, Property Code 2110.
8 Hanford, Volume 1, Design Data, Property Code 3304.

Table 3.3-3 Mechanical Properties of SA 240, Type 304L Stainless Steel

Property (units)	Temperature (°F)							
	-40	-20	+70	+200	+300	+400	+500	+750
Ultimate Strength[1] (ksi)	70.0	70.0	70.0	66.2	60.9	58.5	57.8	55.9
Yield Strength[2] (ksi)	25.0	25.0	25.0	21.4	19.2	17.5	16.4	14.7
Design Stress Intensity[3] (ksi)	16.7	16.7	16.7	16.7	16.7	15.8	14.8	13.3
Modulus of Elasticity[4] (ksi)	28.7E+3	28.7E+3	28.3E+3	27.6E+3	27.0E+3	26.5E+3	25.8E+3	24.4E+3
Alternating Stress[5] @ 10 cycles (ksi)	718.0	718.0	708.0	690.5	675.5	663.0	645.5	610.4
Alternating Stress[5] @ 10^6 cycles (ksi)	28.7	28.7	28.3	27.6	27.0	26.5	25.8	24.4
Coefficient of Thermal Expansion[6] (in/in/°F)	8.13E-6	8.19E-6	8.46E-6	8.79E-6	9.00E-6	9.19E-6	9.37E-6	9.76E-6
Poisson's Ratio[7]	0.275							
Density[8] (lbm/in^3)	0.288							

1 ASME Code, Section II, Part D, Table U.
2 ASME Code, Section II, Part D, Table Y-1.
3 ASME Code, Section II, Part D, Table 2A.
4 ASME Code, Section II, Part D, Table TM-1.
5 ASME Code, Section III, Appendix I, Table I-9.1.
6 ASME Code, Section II, Part D, Table TE-1.
7 Hanford, Volume 1, Design Data, Property Code 2110.
8 Hanford, Volume 1, Design Data, Property Code 3304.

Table 3.3-4 Mechanical Properties of SA 705, SA 693 and SA 564, Type 630, H1150, 17-4 PH Stainless Steel

Property (units)	Temperature (°F)							
	-40	**-20**	**+70**	**+200**	**+300**	**+400**	**+500**	**+650**
Ultimate Strength[1, 2] (ksi)	135.0	135.0	135.0	135.0	135.0	131.4	128.5	125.7[2]
Yield Strength[3] (ksi)	105.0	105.0	105.0	97.1	93.0	89.5	87.0	83.6
Design Stress Intensity[4] (ksi)	45.0	45.0	45.0	45.0	45.0	43.8	42.8	41.9
Modulus of Elasticity[5] (ksi)	28.7E+3	28.7E+3	28.3E+3	27.6E+3	27.0E+3	26.5E+3	25.8E+3	25.1E+3
Alternating Stress[6] @ 10 cycles (ksi)	401.8	401.8	396.2	386.4	378.0	371.0	361.2	341.6
Alternating Stress[6] @ 10^6 cycles (ksi)	19.1	19.1	18.9	18.4	18.0	17.7	17.2	16.3
Coefficient of Thermal Expansion[7] (in/in/°F)	5.88E-6	5.88E-6	5.89E-6	5.90E-6	5.90E-6	5.91E-6	5.91E-6	5.93E-6
Poisson's Ratio[8]	0.291							
Density[9] (lbm/in^3)	0.284							

1 ASME Code, Section II, Part D, Table U.
2 Tabulated value is calculated by ratioing from the Design Stress Intensity.
3 ASME Code, Section II, Part D, Table Y-1.
4 ASME Code, Section II, Part D, Table 2A.
5 ASME Code, Section II, Part D, Table TM-1.
6 ASME Code, Section III, Appendix I, Table I-9.1.
7 ASME Code, Section II, Part D, Table TE-1.
8 ARMCO, Table 7.
9 ARMCO, Table 16.

Table 3.3-5 Mechanical Properties of A36 Carbon Steel

Property (units)	Temperature (°F)							
	100	200	300	400	500	600	650	700
S_u (ksi)[1]	58.0	58.0	58.0	58.0	–	–	–	–
S_y (ksi)[1]	36.0	32.8	31.9	30.8	29.1	26.6	26.1	25.9
S_m (ksi)[2]	19.3	19.3	19.3	19.3	19.3	17.7	17.4	17.3
Modulus of Elasticity[3] (ksi)	29.0E+3	28.8E+3	28.3E+3	27.7E+3	27.3E+3	26.7E+3	26.1E+3	25.5E+3
α (in/in/ °F)[4]	5.53E-6	5.89E-6	6.26E-6	6.61E-6	6.91E-6	7.17E-6	7.30E-6	7.41E-6
Poisson's Ratio[5]	0.31							
Density[6] (lbm/in³)	0.284							

1 Cases of ASME Boiler and Pressure Vessel Code, Case N-71-17, Table 1, 2, 3, 4, 5.
2 ASME Code, Section II, Part D, Table 2A.
3 ASME Code, Section II, Part D, Table TM-1.
4 ASME Code, Section II, Part D, Group C in Table TE-1.
5 ASME Code, Section II, Part D, Table NF-1.
6 Ross.

Table 3.3-6 Mechanical Properties of A615, GR 60, Reinforcing Steel

Property (units)	Temperature (°F)			
	148	**328**	**508**	**688**
S_u (ksi)[1]	58.0	65.3	69.6	53.7
S_y (ksi)[2]	60			
Modulus of Easticity[1] (ksi)	29.88E+3			
α (in/in/ °F)[1]	6.1E-6			
Density[1] (lbm/in^3)	0.284			

1 Ross.
2 Cases of ASME Code, Case N-71-17, Table 1, 2, 3, 4, 5.

Table 3.3-7 Mechanical Properties of A500 Carbon Steel

Property (units)	Temperature (°F)							
	100	200	300	400	500	600	650	700
S_u (ksi)[1]	58.0	58.0	58.0	58.0	–	–	–	–
S_y (ksi)[1]	42.0	38.3	37.2	35.9	33.9	31.0	30.4	30.2
Modulus of Elasticity[2] (ksi)	29.0E+3	28.8E+3	28.3E+3	27.7E+3	27.3E+3	26.7E+3	26.1E+3	25.5E+3
α (in/in/ °F)[3]	5.53E-6	5.89E-6	6.26E-6	6.61E-6	6.91E-6	7.17E-6	7.30E-6	7.41E-6
Poisson's Ratio[4]	0.31							
Density[5](lbm/in³)	0.284							

1 Cases of ASME Boiler and Pressure Vessel Code, Case N-71-17, Table 1, 2, 3, 4, 5.
2 ASME Code, Section II, Part D, Table TM-1.
3 ASME Code, Section II, Part D, Group C in Table TE-1.
4 ASME Code, Section II, Part D, Table NF-1.
5 Ross.

Table 3.3-8 Mechanical Properties of A588, Type A and B Low Alloy Steel

Property (units)	Temperature (°F)							
	100	200	300	400	500	600	650	700
S_u (ksi)[1]	70.0	70.0	70.0	70.0	70.0	70.0	70.0	70.0
S_y (ksi)[1]	50.0	47.5	45.6	43.0	41.8	39.9	38.9	37.9
S_m (ksi)[1]	23.3	23.3	23.3	23.3	23.3	23.3	23.3	23.3
Modulus of Elasticity[2] (ksi)	29.0E+3	28.8E+3	28.3E+3	27.7E+3	27.3E+3	26.7E+3	26.1E+3	25.5E+3
α (in/in/ °F)[3]	5.53E-6	5.89E-6	6.26E-6	6.61E-6	6.91E-6	7.17E-6	7.30E-6	7.41E-6
Poisson's Ratio[4]	0.31							
Density[5] (lbm/in^3)	0.284							

1 Cases of ASME Boiler and Pressure Vessel Code, Case N-71-17, Table 1, 2, 3, 4, 5.
2 ASME Code, Section II, Part D, Table TM-1.
3 ASME Code, Section II, Part D, Group C in Table TE-1.
4 ASME Code, Section II, Part D, Table NF-1.
5 Ross.

Table 3.3-9 Mechanical Properties of 6061-T6 Aluminum Alloy

Property (units)	**Temperature (°F)**							
	70	**100**	**200**	**300**	**400**	**500**	**600**	**700**
Ultimate Strength[1, 2] (ksi)	42.0	40.7	38.2	31.5	17.2	6.7	3.4	2.1
Yield Strength (ksi)[1, 2]	35.0	33.9	32.2	26.9	14.0	5.3	2.5	1.4
Design Stress[1](ksi)	10.5	10.5	10.5	8.4	4.4	N/A	N/A	N/A
Modulus of Elasticity[3](ksi)	10.0E+3	9.9E+3	9.6E+3	9.2E+3	8.7E+3	8.1E+3	7.0E+3[5]	N/A
Coefficient of Thermal Expansion[4] (in/in/°F)	---	12.6E-6	12.9E-6	13.22E-6	13.52E-6	13.7E-6[6]	14.3E-6[6]	N/A
Poisson's Ratio[7]	0.33							
Density[8] (lbm/in^3)	0.098							

1 ASME Code, Section II, Part D, Table 1-B.

2 Strength at elevated temperatures calculated using the following relationships from MIL-HDBK-5G, Figures 3.6.2.2.1(a) and 3.6.2.2.1(b):

$S_{u\ @temp} = (\%\text{Value})\ (S_{u\ @70})$

$S_{y\ @temp} = (\%\text{Value})\ (S_{y\ @70})$

% Value @ Room Temperature

Temp°F	100	200	300	400	500	600	700
S_u	97	91	75	41	16	8	5
S_y	97	92	77	40	15	7	4

3 ASME Code, Section II, Part D, Table TM-2.

4 ASME Code, Section II, Part D, Table TE-2.

5 MIL-HDBK-5G, Figure 3.6.2.2.4.

6 MIL-HDBK-5G, Figure 3.6.2.0.

7 ASME Code, Section II, Part D, Table NF-1.

8 ASME Code, Section II, Part D, Table NF-2.

Table 3.3-10 Mechanical Properties of Concrete

Property (units)	Temperature (°F)					
	70	**100**	**200**	**300**	**400**	**500**
Density[1] (lbs/ft^3)	140	140	140	140	140	140
Compressive Strength[1] (psi)	4,000	4,000	4,000	3,800	3,600	3,400
Coefficient of Thermal Expansion[1] (in/in/°F)	$5.5x10^{-6}$					
Modulus of Elasticity[1] (ksi)	---	$3.64x10^6$	$3.38x10^6$	$3.09x10^6$	$3.73x10^6$	$3.43x10^6$

[1] Fintel.

Table 3.3-11 Mechanical Properties of NS-4-FR

Property (units)	**Temperature (°F)**			
	86	**158**	**212**	**302**
Compressive Modulus of Elasticity[1] (ksi)	561			
Coefficient of Thermal Expansion[1] (in/in/°F)	2.22E-5	4.72E-5	5.88E-5	5.74E-5
Density[1] (lbm/in^3)	0.0607			

[1] GESC Product Data.

Table 3.3-12 Mechanical Properties of SA193, Grade B6, High Alloy Steel Bolting Material

Property (units)	Temperature (°F)							
	-40	-20	70	200	300	400	500	750
S_u (ksi)	110.0[1]	110.0[1]	110.0[1]	104.9[2]	101.5[2]	98.3[2]	95.6[2]	89.4[2]
S_y (ksi)	85.0[1]	85.0[1]	85.0[1]	81.2[2]	78.4[2]	76.0[2]	73.9[2]	69.1[2]
S_m (ksi)[2,3]	28.3	28.3	28.3	27.0	26.1	25.3	24.6	23.0[2]
Modulus of Elasticity[4] (ksi)	30.1E+3	30.1E+3	29.2E+3	28.5E+3	27.9E+3	27.3E+3	26.7E+3	25.2E+3
Alternating Stress[5] @ 10 cycles (ksi)	1,104.4	1,100.0	1,085.0	1,058.0	1,035.0	1,015.0	989.0	935.3
Alternating Stress[5] @ 10^6 cycles (ksi)	13.0	12.9	12.7	12.4	12.2	11.9	11.6	11.0
α (in/in/ °F)[6]	5.73E-6	5.76E-6	5.92E-6	6.15E-6	6.30E-6	6.40E-6	6.48E-6	6.64E-6
Density[7] (lbm/in^3)	0.283							

1 ASME Code, Section III, Appendix 1, Table I-1.3.
2 Tabulated values are calculated by ratioing from the Design Stress Intensity.
3 ASME Code, Section II, Part D, Table 4.
4 ASME Code, Section II, Part D, Table TM-1.
5 ASME Code, Section III, Appendix 1, Table I-9.1.
6 ASME Code, Section II, Part D, Table TE-1.
7 Baumeister and Marks, "Standard Handbook for Mechanical Engineers," 1966.

Table 3.3-13 Mechanical Properties of SA-350/A-350, Grade LF 2, Class 1 Low Alloy Steel

Property	Value					
Temperature (°F)	70	200	300	400	500	700
Ultimate strength, [1] S_u, (ksi)	70.0	70.0	70.0	70.0	70.0	70.0
Yield strength, [1] S_y (ksi)	36.0	32.8	31.9	30.8	29.1	25.9
Design Stress Intensity, [1] S_m (ksi)	23.3	21.9	21.3	20.6	19.4	17.3
Modulus of Elasticity, [1] E, ($\times 10^3$ ksi)	29.2	28.5	28.0	27.4	27.0	25.3
Coefficient of Thermal Expansion, [1] α ($\times 10^{-6}$ in/in/°F)	—	5.89	6.26	6.61	6.91	7.41
Alternating Stress [2] at 10^6 cycles (ksi)	12.5	12.2	11.9	11.7	11.5	10.8
Alternating Stress [2] at 10 cycles (ksi)	580.0	566.0	556.1	544.2	536.3	502.5
Poisson's Ratio [1]	0.31					
Density [1]	0.279 lbm/in^3					

1 ASME Code, Section II, Part D.

2 ASME Code, Appendix I

3.4 General Standards for Casks

3.4.1 Chemical and Galvanic Reactions

The materials used in the fabrication and operation of the NAC-MPC system have been evaluated to determine whether chemical, galvanic, or other reactions among the materials, contents, and environments can occur. All phases of operation—loading, unloading, handling, and storage—have been considered for the environments that may be encountered under normal, off-normal, or accident conditions. Based on the evaluation, there is one potential reaction that could adversely affect the overall integrity of the storage cask, the fuel basket, the transportable storage canister, or the structural integrity and retrievability of the fuel from the canister. That potential reaction, between aluminum and spent fuel pool water, which may produce hydrogen is mitigated by the specific canister loading procedures presented in Section 8.1.1.

3.4.1.1 Component Operating Environment

Most of the component materials of the NAC-MPC are exposed to two typical operating environments: 1) an open canister containing pool water or borated water with a pH of 4.5 and spent fuel or other radioactive material; or 2) a sealed canister containing helium, but with the canister in environs that include air, rain water/snow/ice, and marine (salty) water/air. The spent fuel assemblies consist of zircaloy or stainless steel clad fuel and other fuel assembly components of stainless steel.

Each category of canister component materials is evaluated for potential reactions in each of the operating environments to which those materials are exposed. These environments may occur during fuel loading or unloading, handling or storage, and include normal, off-normal, and accident conditions.

One of the operating environments to which the canister internal component materials are exposed does not provide the conditions necessary for a reaction (corrosion); i.e., both moisture and oxygen must be present for corrosion to occur. This long-term environment is the sealed canister, backfilled with helium. Helium displaces the oxygen in the canister effectively precluding corrosion. Galvanic corrosion (i.e., between dissimilar metals that are in contact) could occur, but only if there is water present at the point of contact and the metals are in electrical contact with each other (i.e., mechanically held together). NAC's operating procedures provide two helium backfill cycles in series separated by a vacuum-drying cycle for the canister during the preparation of the canister for storage. Therefore, the canister cavity is effectively dry and galvanic corrosion is precluded.

3.4.1.2 Component Material Categories

The component materials evaluated are categorized based on similarity of physical and chemical properties and/or on similarity of component functions. The categories of materials that are considered are stainless/nickel alloy steels, nonferrous metals, and criticality control materials. These categories are evaluated based on the environment to which they could be exposed during operation or use of the canister.

The canister component materials are not reactive among themselves, with the canister's contents, nor with the canister's operating environments, except aluminum, during any phase of normal, off-normal, or accident condition loading, unloading, handling, or storage operations. Therefore, only the potential aluminum reaction with spent fuel pool water is evaluated.

3.4.1.2.1 Stainless Steels

No reaction of the canister component stainless steel is expected in any environment except for the marine environment, where chloride-containing salt spray might initiate pitting of the steels if the chlorides are allowed to concentrate and stay wet for extended periods of time (weeks). Only the external canister surface could be so exposed. The corrosion rate will, however, be so low that no detectable corrosion products or gases will be generated. The NAC-MPC has smooth external surfaces to minimize the collection of such materials as salts.

There is a significant electrochemical potential difference between austenitic (300 series) stainless steel and aluminum. If aluminum is in electrical contact with the austenitic stainless steel, the aluminum could be expected to exhibit corrosion driven by electrochemical EMF when immersed in water. Pressurized water reactor (PWR) pool water does provide a conductive potential. The only aluminum components that will be in contact with stainless steel and exposed to the pool water are the heat transfer disks in the fuel basket. Since the fuel basket is not welded or bolted to the canisters, poor, if any, electrical contact with the stainless steels is present. Therefore, in the absence of electrical contact between the aluminum and the stainless steel, galvanically driven corrosion does not occur.

No coatings are applied to the stainless steels.

Based on the foregoing discussion, there are no potential reactions associated with the stainless steel canister components.

3.4.1.2.2 Nonferrous Metals

Heat transfer disks fabricated from 6061-T6 aluminum alloy are used in the NAC-MPC fuel basket to augment heat transfer from the spent fuel through the basket structure to the canister exterior. Vendor and Nuclear Regulatory Commission safety evaluations of the NUHOMS Dry Spent Fuel Storage System (Docket No. 72-1004) have concluded that combustible gases, primarily hydrogen, may be produced by a chemical reaction and/or radiolysis when aluminum or aluminum flame-sprayed components are immersed in spent fuel pool water. The evaluations further concluded that it is possible, at higher temperatures (above 150°F to 160 °F), for the aluminum/water reaction to produce a hydrogen concentration in the cask or canister that approaches or exceeds the Lower Flammability Limit (LFL) for hydrogen of 4 percent. The NRC Inspection Reports No. 50-266/96005 and 50-301/96005 dated July 01, 1996, for the Point Beach Nuclear Plant concluded that hydrogen generation by radiolysis was insignificant relative to other sources.

Thus, it is reasonable to conclude that small amounts of combustible gases, primarily hydrogen, may be produced during NAC-MPC canister loading or unloading operations as a result of a chemical reaction between the 6061-T6 aluminum heat transfer disks in the fuel basket and the spent fuel pool water. The generation of combustible gases stops when the water is removed from the cask or canister and the aluminum surfaces are dry.

A galvanic reaction may occur at the contact surfaces between the aluminum disks and the stainless steel tie rods and spacers in the presence of an electrolyte, like the pool water. The galvanic reaction ceases when the electrolyte is removed. Each metal has some tendency to ionize, or release electrons. A voltage, or electromotive force (emf), associated with this release of electrons is generated between two dissimilar metals in an electrolytic solution. The emf between aluminum and stainless steel is small and the amount of corrosion is directly proportional to the emf. Loading operations generally take less than 24 hours, a large portion of which has the canister immersed in and open to the pool water after which the electrolyte (water) is drained and the cask or canister is dried and back-filled with helium, effectively halting any galvanic reaction.

The potential chemical or galvanic reactions do not have a significant detrimental effect on the ability of the aluminum heat transfer disks to perform their function for all normal and accident conditions associated with dry storage.

Loading Operations

After the canister is removed from the pool and during canister closure operations, an air space is created inside the canister beneath the shield lid by the drain-down of 50 gallons of water so that the shield-lid-to-canister-shell can be performed. The resulting air space is approximately 70 inches in diameter and 3 inches deep. Although there is some clearance between the inside diameter of the canister shell and the outside diameter of the shield lid, it is possible that gases released from a chemical reaction inside the canister could accumulate beneath the shield lid. A bare aluminum surface oxidizes when exposed to air, reacts chemically in an aqueous solution and may react galvanically when in contact with stainless steel in the presence of an aqueous solution.

The reaction of aluminum in water, which results in hydrogen generation, proceeds as:

$$2\ Al + 3\ H_20 \Rightarrow Al_2O_3 + 3\ H_2$$

The aluminum oxide (Al_2O_3) produces the dull, light gray film that is present on the surface of bare aluminum when it reacts with the oxygen in air or water. The formation of the thin oxide film is a self limiting reaction as the film isolates the aluminum metal from the oxygen source acting as a barrier to further oxidation. The oxide film is stable in pH neutral (passive) solutions, but is soluble in borated PWR spent fuel pool water. The oxide film dissolves at a rate dependent upon the pH of the water, the exposure time of the aluminum in the water, and the temperatures of the aluminum and water.

PWR spent fuel pool water is a boric acid and demineralized water solution. The pH, water chemistry, and water temperature vary from pool to pool. Since the reaction rate is largely dependent upon these variables, it may vary considerably from pool to pool. Thus, the generation rate of combustible gas (hydrogen) that could be considered representative of spent fuel pools in general is very difficult to accurately calculate.

To ensure safe loading and/or unloading of the NAC-MPC transportable storage canister, the loading and unloading procedures defined in Chapter 8 provide for the monitoring of hydrogen gas before and during the welding operations joining the shield lid to the canister shell and joining the vent and drain port coverplates to the shield lid. The monitoring system shall be capable of detecting hydrogen at 60% of the lower flammability limit for hydrogen (i.e., 0.6 x 4.0 = 2.4%). The hydrogen detector shall be mounted so as to detect hydrogen prior to initiation of the weld, and continuously at a point ahead of the weld head. Detection of hydrogen in a concentration exceeding 2.4% shall be cause for the welding operation to stop. If hydrogen gas is detected at concentrations above 2.4% at any time, the hydrogen gas shall be removed by flushing ambient air into the region below the shield lid or port coverplate. To remove hydrogen from below the shield lid, the vacuum pump is attached to the vent port and operated for a sufficient period of time to remove at least five times the air volume of the space below the lid by drawing ambient air through the gap between the shield lid and the canister shell, thus removing or diluting any combustible gas concentrations.

The vacuum pump shall exhaust to a system or area where hydrogen flammability is not an issue. If hydrogen gas is detected at the port coverplates, the coverplate is removed and service air is used to flush combustible gases from the port. Once the root pass weld is completed, there is no further likelihood of a combustible gas burn because the ignition source is isolated from the combustible gas. Once welding of the shield lid has been completed, the canister is drained, vacuum dried and back-filled with helium.

No hydrogen is expected to be detected prior to, or during, the welding operations. The vent port in the shield lid remains open from the time that the loaded canister is removed from the spent fuel pool until the time that the vent port coverplate is ready to be welded to the shield lid. Since the postulated combustible gases are very light, the open vent port provides an escape path for any gases that are generated prior to the time that the canister is vacuum dried. Once the canister is dry, no combustible gases form within the canister. The mating surfaces of the support ring and inner lid are machined to provide a good level fitup, but are not machined to provide a metal to metal seal. Consequently, additional exit paths for the combustible gases exist at the circumference of the shield lid.

Unloading Operations

It is not expected that the canister will contain a measurable quantity of combustible gases during the time period of storage. The canister is vacuum dried and backfilled with helium immediately prior to being welded closed. There are only minor mechanisms by which hydrogen is generated after the canister is dried and sealed.

As shown in Section 8.3, the principal steps in opening the canister are the removal of the structural lid, the removal of the vent and drain port coverplates, and the removal of the shield lid. These steps are performed by cutting or grinding. The design of the canister precludes monitoring for the presence of combustible gases prior to the removal of the structural lid and the vent and drain port coverplates. Following removal of the vent port coverplate, a vent line is connected to the vent port quick disconnect. The vent line incorporates a hydrogen gas detector, which is capable of detecting hydrogen at a concentration of 2.4% (60% of its lower flammability limit of 4%). The pressurized gases (expected to be greater than 96% helium) in the canister are expected to carry combustible gases out of the vent port. If the exiting gases in the vent line contain no hydrogen at concentrations above 2.4%, the drain port coverplate weld is cut and the coverplate removed. If levels of hydrogen gas above 2.4% concentration are detected in the vent line, then the vacuum system is used to remove all residual gas prior to removal of the drain port coverplate. During the removal of the drain port coverplate, the hydrogen gas detector is attached to the vent port to ensure that the hydrogen gas concentrations remains below 2.4%. Following removal of the drain port coverplate, the canister is filled with water using the vent and drain ports. Prior to cutting the shield lid weld, 50 gallons of water are removed from the canister to permit the removal of the shield lid. Monitoring for hydrogen would then proceed as described for the loading operations.

3.4.1.2.3 Criticality Control Material

The criticality control material is a sheet consisting of boron carbide mixed in an aluminum alloy. This material is effectively a sheet of aluminum that is in contact with the stainless steel fuel tubes and is completely enclosed by a welded stainless steel cover. This "aluminum" is protected by an oxide layer that forms shortly after fabrication and precludes further oxidation of the aluminum or interaction with the stainless steel. Consequently, there are no potential reactions associated with the aluminum-based criticality control material.

3.4.1.2.4 Coatings

The exposed carbon steel surfaces of the transfer cask are coated with Keeler & Long E-Series epoxy enamel. Exposed carbon steel in the vertical concrete cask and transfer cask adapter plate may be coated with either E-Series epoxy enamel or Ameron PSX 738 Siloxane coating. E-Series epoxy enamel is approved for use in Coating Service Level 1 areas of nuclear power plants, which include the spent fuel pool and reactor containment areas. These are chemically resistant, shop-applied coatings that are designed for elevated service temperatures. Neither of the coatings contains Zinc, nor do they offgas when exposed to borated water. The coatings also serve to chemically isolate the carbon steel from stainless steel and aluminum components used in the NAC-MPC. The technical specifications for Keeler & Long E-Series epoxy enamel and for Ameron PSX 738 Siloxane coating are provided in Sections 3.8.1 and 3.8.2, respectively.

3.4.1.3 General Effects of Identified Reactions

One potential chemical, galvanic, or other reaction has been identified for the NAC-MPC system. A condition, such as the generation of flammable or explosive quantities of combustible gases does not result in an adverse event during any phase of canister operations for normal, off-normal, or accident conditions because specific detection procedures and responses are defined in Section 8.1.1.

3.4.1.4 Adequacy of the Canister Operating Procedures

Based on this evaluation, which resulted in one identified reaction, it is concluded that the NAC-MPC operating controls and procedures presented in Chapter 8.0 are adequate to minimize the occurrence of hazardous conditions.

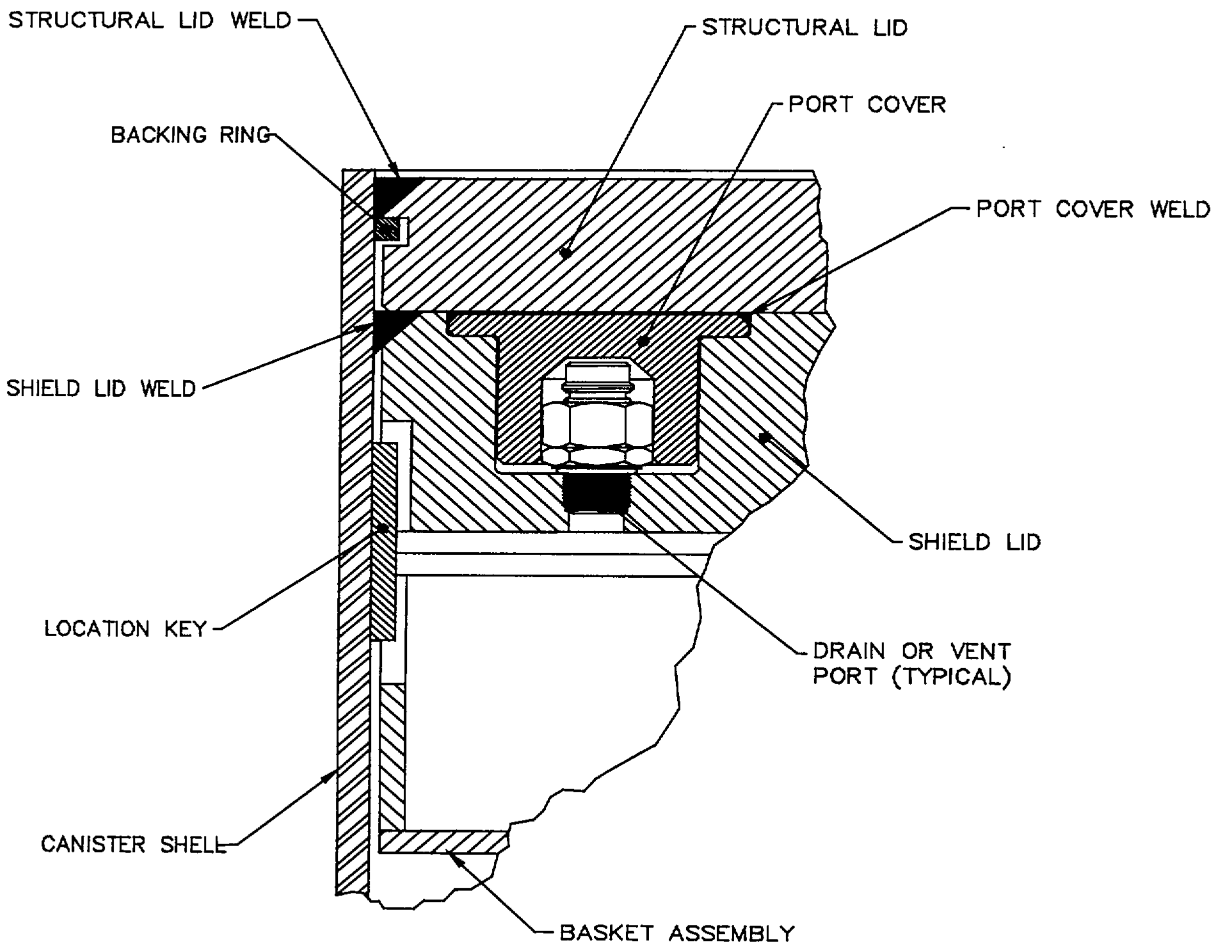
3.4.1.5 Effects of Reaction Products

One potential chemical, galvanic, or other reaction has been identified for the NAC-MPC. The effects of that reaction are mitigated by the operating procedures described in Section 8.1.1. The overall integrity of the canister and the structural integrity and retrievability of the spent fuel are not adversely affected for the design basis life of the canister. Based on the evaluation, there will be no change in the canister or fuel cladding thermal properties, there will be no binding of mechanical surfaces, no change in basket clearances, and no degradation of any safety components either directly or indirectly.

3.4.2 Positive Closure

The NAC-MPC employs a positive closure system that is composed of multipass welds to join the canister shield lid to the shell, and to join the canister structural lid to the shell. Port covers that are welded to the shield lid (see Figure 3.4.2-1) close the penetrations to the canister cavity through the shield lid. The welds employed for closing the NAC-MPC canister preclude inadvertent opening of the canister. A bolted lid closes the top of the concrete storage cask. The lid weighs approximately 3,000 pounds. The weight of the lid, its inaccessibility and the presence of the bolts effectively preclude inadvertent opening of the lid.

Figure 3.4.2-1 NAC-MPC Welded Closure System



3.4.3 Lifting Devices

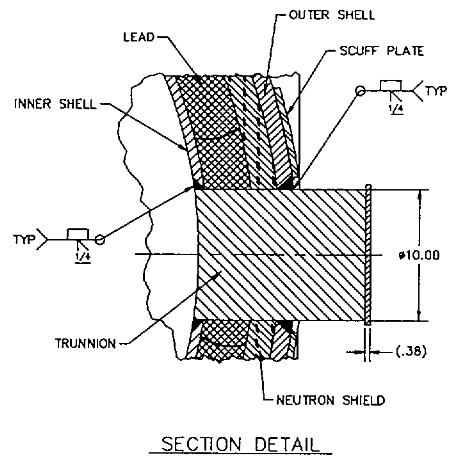
To provide more efficient handling of the components of the NAC-MPC system, different methods of lifting, i.e., trunnions, hoist rings, and jacks and air pads, have been designed for each of the components—the transfer cask, the transportable storage canister, and the storage cask, respectively.

The transfer cask is lifted by two trunnions located near the top of the cask. The 10-inch diameter trunnions extend through the multiwall body to 5.25 inches beyond the outer shell and are full-penetration welded to both the inner and the outer shells (Figure 3.4.3-1). The transfer cask is designed as a heavy-lifting device that satisfies the requirements of NUREG-0612 and ANSI N14.6 for lifting the combined weight of the transfer cask and a fully loaded canister of fuel and water (approximately 143,000 pounds). This is the maximum weight for the transfer cask during a lifting operation.

The transportable storage canister is lifted and handled, while supported on the shield doors within the transfer cask during all preparation, loading and cask closure operations and is then moved to the top of the storage cask. Six hoist rings that are threaded into the structural lid are used to lift the loaded and closed canister just off the shield doors of the transfer cask and to lower the canister into the storage cask after the shield doors are opened. The hoist rings are also used for any subsequent lifting of the loaded canister whose weight is approximately 54,730 pounds (Figure 3.4.3-2).

The vertical concrete cask is raised approximately 3 inches by four lifting jacks placed at the jacking pads located near the end of each air inlet. A system of air pads consisting of 4 units is then inserted under the concrete cask. The cask is lowered onto the air pads (uninflated) and the jacks are removed. The air pads are inflated to lift the concrete cask and position it as required on the storage pad or on the transport vehicle. When positioning is complete, the jacks are again used to raise the cask and remove the air pads.

Figure 3.4.3-1 Transfer Cask Lifting Trunnion Design



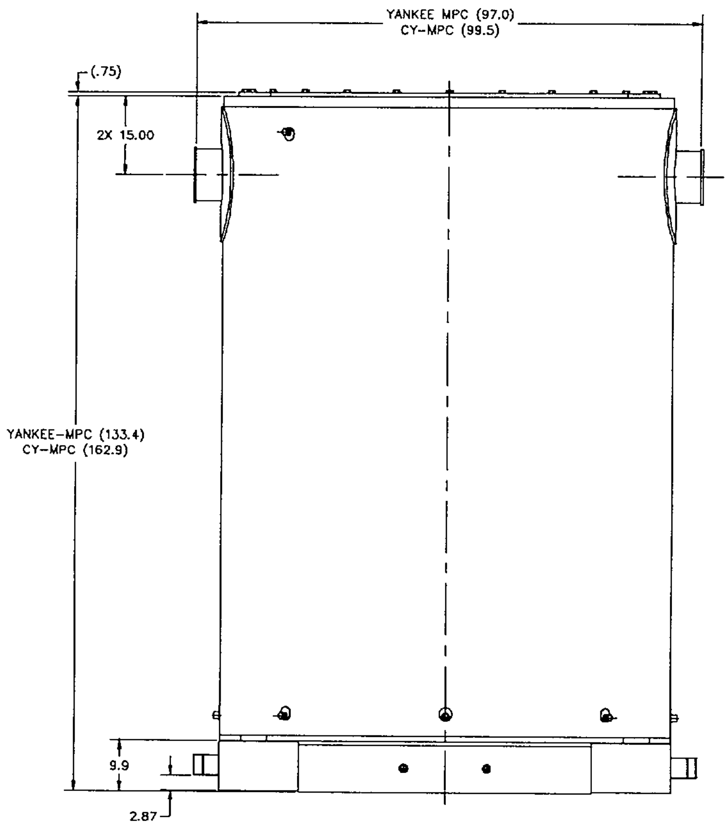
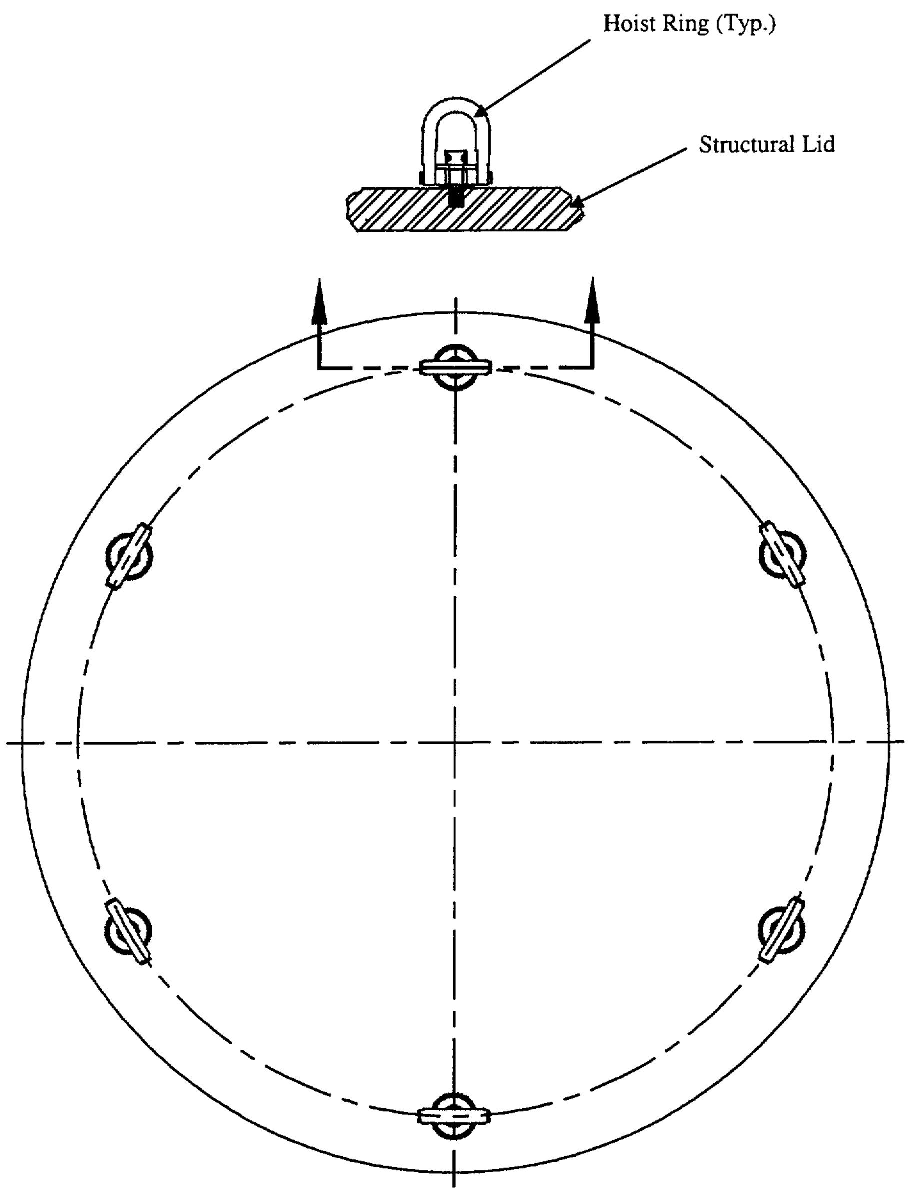


Figure 3.4.3-2 Canister Hoist Ring Design



3.4.3.1 Storage Cask Bottom Lift

The concrete cask is lifted from the bottom via an air pad system. Insertion of the air pad system is made possible by the use of lifting jacks. The current design utilizes a Synchronous Lifting System with four (4) hydraulic jack cylinders. The system is designed to equally distributes hydraulic pressure among four cylinders.

3.4.3.1.1 Bottom Lift By Hydraulic Jack

To ensure the concrete bearing stresses at the jack locations do exceed the allowable stress, the required diameter of the jack piston rod is determined. The concrete allowable bearing capacity (in pounds) at each jack is:

$$U_b = \phi f_c{}'A = \frac{(0.7)(4{,}000)\pi d^2}{4} = 2{,}199.1 d^2,$$

where:

ϕ = 0.7 strength reduction factor for bearing,

f_c' = 4,000 psi concrete compressive strength,

$A = \frac{\pi d^2}{4}$, concrete bearing area (d = bearing area diameter).

For dead load conditions, the concrete bearing strength must be greater than the storage cask weight times a load factor, L_f, of 1.4:

$$2{,}199.1\, d^2 > \frac{L_f \times W}{n} = \frac{1.4(206{,}100)}{4} \Rightarrow d > 5.73 \text{ in.},$$

where:

n = the number of jacks, 4

W = the weight of the VCC, 206,100 lbs

Lf = the load factor, 1.4

The diameter obtained in the above equation is the maximum permissible area at the surface of the concrete over which the load must be distributed. The force exerted by the jack to lift the storage cask is applied to the bottom surface of the top plate of the air inlet, which is separated from the concrete surface by a one inch thick steel plate. The force exerted by the jack will be distributed over a larger area on the concrete surface. The effective diameter of the load acting on the concrete surface is increased by 2 x tan (45°) x thickness of the steel plate. Therefore, the required hydraulic jack piston diameter is:

5.73 in. - 2 in. = 3.73 in.

The actual hydraulic cylinder to be used has a piston rod diameter of 4 1/8 inch, which is greater than the required 3.73 inch.

Nelson Studs

During the bottom lift of the storage cask with hydraulic jacks, the weight of the loaded canister, pedestal and air inlet vent system are transferred to the baseplate of the storage cask (total weight = 63,230 lbs). As the baseplate is loaded, the plate tends to flex, thus separating the concrete from the bottom plate. To prevent this separation, Nelson studs are utilized to bond the concrete to the bottom plate.

Use of the Nelson studs requires proper spacing to prevent overlapping of the shear cones of adjacent Nelson studs (TRW). The term "shear cone" refers to the geometry that a failed concrete section takes when a Nelson stud pulls out of concrete. Overlapping of adjacent shear cones tends to reduce the effectiveness of the Nelson stud. In the case of the storage cask, thirty-two (32), 3/4 × 6 3/16 inch Nelson studs are used. For 4,000 psi strength concrete, the minimum spacing between two studs is 7.984 inches (2 x 3.992 inch, where, 3.992 inch is surface area diameter of the full concrete shear cone for 3/4 x 6-3/16 inch Nelson studs.). The spacing of the Nelson studs on the storage cask bottom exceeds requirements. The total capacity is:

$$\text{Capacity} = 32 \ \text{Anchors} \times 23{,}860 \ \text{lb}/\text{Anchor} = 763{,}500 \ \text{lb}$$

The allowable load, P_u, with a load factor of 2.0, as specified in the TRW design data, is:

$$P_u = \frac{763{,}500}{2.0} = 381{,}750 \, \text{lb}$$

The total load applied to the storage cask bottom plate (including a 10% dynamic load factor) is:

$$63{,}230 \times 1.1 = 69{,}553 \ \text{lbs}$$

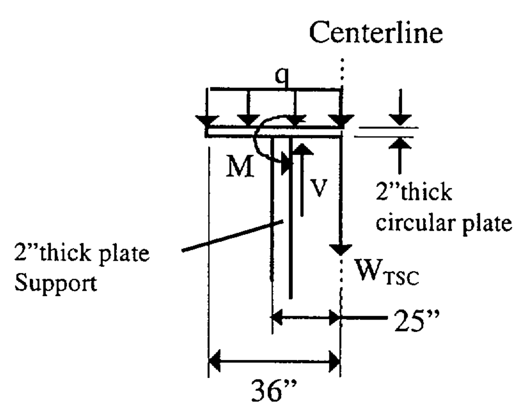
Therefore, the margin of safety is:

$$\text{M.S.} = \frac{381{,}750}{69{,}553} - 1 = +4.49$$

Pedestal - Horizontal Plate

The canister weight (54,730 pounds) is uniformly distributed over the 2-inch thick horizontal circular plate of the pedestal. The self-weight of the plate is 2,313 pounds (0.284 lbs/in^3 x π x 36^2 x 2). The equivalent pressure is:

$$q = \frac{W}{A} = \frac{54{,}730 + 2{,}313}{\pi(36^2)} = 14.01 \ \text{psi}$$



The horizontal plate is supported by the pedestal ring (vertical plate support) at four locations (each has a circumferential length of 16.85 inch). The bending moment at the cross-section of the plate at support location is:

$$M = (LF)(q)(A)(P_c) = (1.1)(14.01)\left(\frac{\pi(36^2 - 25^2)}{4}\right)(5.5) = 44{,}670 \text{ in.-lb},$$

where:

A = the area of the pedestal from the plate support to the edge of the circular plate,
P_c = 5.5 in., the location of the resultant force,
LF = 1.1, load factor to account for 10% dynamic load factor

The bending stress is:

$$f_b = \frac{6M}{bt^2} = \frac{(6)(44{,}670)}{(16.85)(2)^2} = 3{,}977 \text{ psi}$$

The material of the pedestal horizontal plate is ASTM A36 with a yield stress (Fy) of 36,000 psi. The allowable stress for flexural members per the Manual of Steel Construction (AISC) is:

$$F_b = 0.66\ F_y = 23{,}760 \text{ psi},$$

and the resulting margin of safety is:

$$M.S. = \frac{23{,}760}{3{,}977} - 1 = +5.0$$

The maximum shear stress at the support location is:

$$f_v = \frac{W}{L} = \frac{(54{,}730 + 2{,}313)(1.1)}{4(16.85)(2)} = 465.5 \text{ psi}$$

The allowable shear stress is 14,400 psi (0.4 F_y = 0.4 x 36,000) and the margin of safety is:

$$M.S. = \frac{14{,}400}{465.5} - 1 = +29.9$$

Pedestal Ring (Vertical Plate)

The pedestal ring is subjected to an axial compressive force and bending moments, due to weight of the canister (54,730 lbs), weight of the pedestal horizontal plate (2,313 lbs) and self-weight of the pedestal ring (0.284 lbs/in. × 16.85 × 2 × 6 = 230 lbs). The maximum compressive stress at the critical cross-section (2 inch × 1.5 inch, 8 locations) is:

$$f_a = \frac{(54{,}730 + 2{,}313 + 230)(1.1)}{8(1.5)(2)} = 2{,}625 \text{ psi.}$$

The allowable stress, F_a, for compression member is determined per Chapter E of the Manual of Steel Construction (AISC):

$$\frac{KL}{r} = \frac{0.65 \times 6}{0.433} = 9,$$

where:

K = 0.65, effective-length factor for the end conditions (rotation and translation fixed),

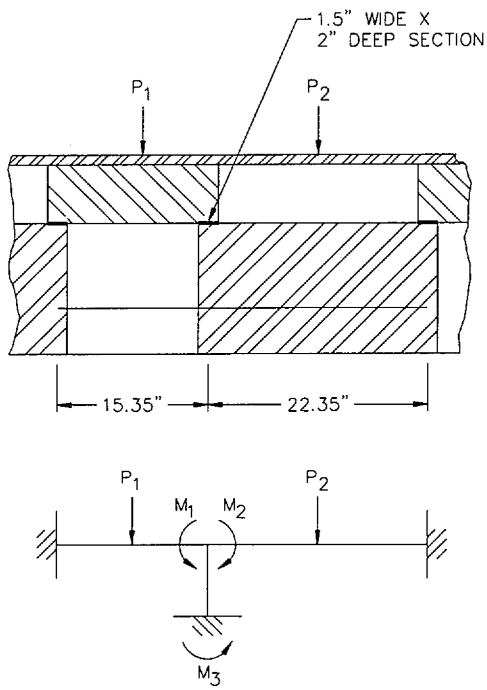
L = 6.0 in., height of pedestal ring (unbraced),

r = $\frac{1.5}{\sqrt{12}} = 0.433$, radius of gyration.

Using Chapter E and Tables 3 through 5 of Section 5 of the AISC for A36 material (F_y = 36,000 psi), the allowable stress, F_a, for compression is determined to be 21,200 psi.

The bending stress at the same cross-section is conservatively calculated below:

P_t = one-fourth of the total load = (54,730 + 2,313) / 4 = 14,261 lbs ≈ 14,300 lbs



The pedestal is represented as a combination of beams to describe the load path.

P_1 = 14,300 × (15.35 / 37.7) = 5,822 lbs

P_2 = 14,300 × (22.35 / 37.7) = 8,478 lbs

M_1 and M_2 are conservatively considered to be the fixed-end moments of beams with a concentrated load at mid-span (with a 10% dynamic load factor). L_1 (15.35 inch) and L_2 (22.35 inch) are the length of the beams. M_3 (the moment at the 2 inch by 1.5 inch cross-section) is considered to be the difference of M_1 and M_2.

$$M_1 = \frac{1.1(P_1 L_1)}{8} = \frac{(1.1)(5,822)(15.35)}{8} = 12,288 \text{ in.} - \text{lbs}$$

$$M_2 = \frac{1.1(P_2L_2)}{8} = \frac{(1.1)(8{,}478)(22.35)}{8} = 26{,}054 \text{in.} - \text{lbs}$$

$$M_3 = M_2 - M_1 = 13{,}766 \text{ in-lbs}$$

The maximum bending stress f_b is computed as:

$$f_b = \frac{6M_3}{bt^2} = \frac{(6)(13{,}766)}{(2)(1.5)^2} = 18{,}355 \text{psi}$$

The allowable stress for bending (F_b) is 23,760 psi (0.66 F_y). Since f_a/F_a is less than 0.15, Equation (H1-3) in the Manual of Steel Construction (AISC), Chapter H, is used to evaluate combined stress:

$$\frac{f_a}{F_a} + \frac{f_b}{F_b} = \frac{2{,}625}{21{,}200} + \frac{18{,}355}{23{,}760} = 0.90 < 1.0$$

Therefore, the pedestal is structurally adequate to support the weight of the loaded canister.

3.4.3.1.2 Bottom Support by Air Pads

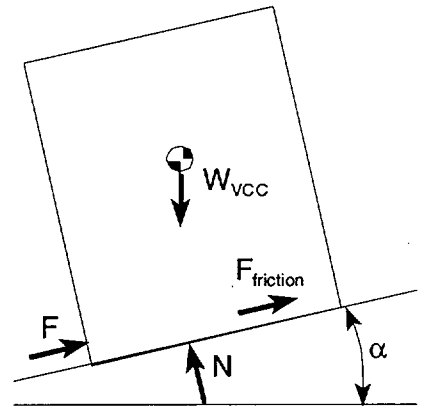
The storage cask is supported by air pads in each of 4 quadrants during transport. The layout of the air pads (four 48 in. × 48 in. square pads) must clear the air inlet locations by approximately 3 inches to allow for hydraulic jack access.

The air pad system maximum lift elevation to the storage cask bottom surface is 5.69 inches (3-inch lift, maximum, plus 2-11/16-inch overall height, when deflated) above the transport surface. The air pad system has a vendor rated lift capacity of 360,000 pounds.

The air pad system must supply sufficient force to overcome the weight of the storage cask under full load with a lift load factor, L.F of 1.1. Assuming a fully loaded weight of 206,100 pounds: the required lift load is 1.1(206,100) = 226,710 pounds. Since the available lift force is greater than the required lift force, the air pads are adequate to lift the storage cask. The lifting force margin of safety = (360,000 / 226,700) -1 = +0.59.

Operational/Handling Requirements

The handling force required to control the storage cask on a sloped surface, assuming minimal friction ($F_{friction}$), is:



$$F = W_{VCC} \sin \alpha$$

$$= 206{,}100 \sin \alpha$$

where W_{VCC} is the weight of the storage cask.

The amount of handling force, F, required on slanted surfaces can be very high. Therefore, the storage cask, while supported by air pads, will be transferred from trailer deck to storage pad site over essentially flat horizontal surfaces.

Once positioned on the storage pad, the storage cask is lifted from the air pads by the jacks for air pad removal. The maximum air pad deflated height plus lift height during all handling and transport is 5.69 inches above the transporter or storage pad surface. Therefore, the storage pad handling method limits the potential drop height to 5.69 inches.

3.4.3.2 Canister Lift

The adequacy of the canister lifting devices is demonstrated by considering each of the hoist rings, the canister structural lid, and the weld that joins the structural lid to the canister shell. Lifting of the canister employs redundant three-legged lifting slings for single failure proof lifting in accordance with NUREG-0612.

When considering a three-point lifting configuration, the load-bearing members of the canister maintain a factor of safety greater than three based on material yield strength. Additionally, the load-bearing members of the canister maintain a factor of safety greater than five based on material ultimate strength. The hoist rings are designed with a 5 to 1 safety factor based on ultimate. Therefore, the lifting requirements are satisfied.

Each lifting device in a three-point lift would experience the following load, F, when lifting the canister (the total load is based on the dead weight of the loaded canister with a dynamic load factor of 10 %):

$$F = \frac{54{,}730 \text{x} 1.1}{3} = 20{,}068 \text{ lbs}$$

The hoist rings used to lift the canister have a rated load capacity of 24,000 pounds with a 5 to 1 safety factor based on material ultimate strength. The length of the hoist ring bolt is 2.50 inches. The following calculation (using a formula from Machinery's Handbook) demonstrates that a thread engagement length of 2.50 inches is satisfactory.

Definition of Terms

D	= basic major diameter of bolt threads = 1.5 inches
n	= number of threads per inch = 6
A_t	= tensile area of the bolt thread (in^2)
L_e	= minimum thread engagement length for mating materials with equal tensile strengths (in.)
$K_n max$	= maximum minor diameter of lid (internal) thread = 1.35 inches
$E_s min$	= minimum pitch diameter of bolt (external) threads = 1.3812 inches
A_s	= shear area of bolt threads (in^2)
A_n	= shear area of lid threads (in^2)
$D_s min$	= minimum major diameter of bolt (external) threads = 1.4794 inches
$E_n max$	= maximum pitch diameter of lid (internal) threads = 1.402 inches
Q	= length of thread engagement required to prevent shearing when the mating thread materials are different tensile strengths (in.)
J	= scale factor for calculation of Q
H	= height of sharp "V" thread = 0.1443 inches

Hoist ring material	= 4140 High strength alloy steel
Tensile strength of 4140	= 180,000 psi
Hoist ring thread	= 1-1/2 - 6 UNC
Structural lid material	= Type 304 L stainless steel
Temperature of structural lid	= 250°F
Tapped hole in structural lid	= 1-1/2 - 6 UNC

For steels up to 100,000-psi tensile strength, the tensile area of the bolt thread is given by:

$$A_t = 0.7854\left(D - \frac{0.9743}{n}\right)^2$$

$$A_t = 1.405 \text{ in}^2$$

For mating materials having equal tensile strengths, the minimum length of thread engagement is given by:

$$L_e = \frac{2A_t}{3.1416 K_n \max\left[\frac{1}{2} + 0.57735 n \left(E_s \min - K_n \max\right)\right]}$$

$$L_e = 1.0898 \text{ inches}$$

For mating materials of differing tensile strengths:

$$J = \frac{A_s \text{ x tensile strength of bolt thread material}}{A_n \text{ x tensile strength of lid thread material}}$$

The shear areas are calculated as follows:

$$A_s = 3.1416 n L_e K_n \max\left[\frac{1}{2n} + 0.57735\left(E_s \min - K_n \max\right)\right]$$

$$A_s = 2.8106 \text{ in}^2$$

$$A_n = 3.1416 n L_e D_s \min\left[\frac{1}{2n} + 0.57735\left(D_s \min - E_n \max\right)\right]$$

$$A_n = 3.8871 \text{ in}^2$$

At a temperature of 250°F, the ultimate strength of Type 304L stainless steel is 63,550 psi.

$$J = \frac{A_s \text{ x} \left(S_u\right)_{bolt}}{A_n \text{ x} \left(S_u\right)_{lid}}$$

$$= 2.048$$

The length of thread engagement necessary to prevent shearing is:

$$Q = JL_e$$

$$Q = (2{,}048)(1.073) = 2.20\text{in.} < 2.5\text{in.}$$

Therefore, the required length of thread engagement is 2.23 inches, which is less than the hoist ring thread length of 2.5 inches.

The hoist rings are rated at 24,000 lbs with a safety factor of 5. The ultimate capacity of the rings is, therefore, 24,000 lbs x 5 = 120,000 lbs (120 kip).

Because the hoist rings and sling legs must demonstrate a safety factor of 10 when compared to ultimate strength, a design load, F_D = 11.0 kips, is chosen.

$$\frac{\text{Ring Ultimate Capacity}}{\text{Design Load}} = \frac{120 \text{ kip}}{11.0 \text{ kip}} = 10.9 > 10$$

The design load, F_D, is:

$$F_D = \sqrt{F_H^2 + F_V^2}$$

where:

$F_H = (F_V)(\tan\theta)$	horizontal load on the hoist ring, where θ is the angle of the sling with respect to the vertical axis of the canister
$F_V = 10.0$ kip	vertical load on the hoist ring

Given a six-point lift, and applying a dynamic load factor of 10%, each hoist ring sees a vertical load, F_V = 1.1 (54,730 lbs)/6 = 10.0 kip. Since

$$F_D = \sqrt{F_H^2 + F_V^2} = F_D = \sqrt{(F_V \tan\theta)^2 + F_V^2}$$

Solving for θ:

$$\theta = \tan^{-1} \frac{\sqrt{(11.0\text{kips})^2 - (10.0\text{kips})^2}}{10.0\text{kip}} = 24.6 \text{ degrees}$$

The distance from the center line of the canister to the hoist ring bolt circle is 30.25 inches. The minimum allowable distance, Y, from the top of the canister is:

$$Y = \frac{X}{\tan\theta} = \frac{30.25}{\tan 24.6} = 66.1 \text{ inches}$$

Therefore, the minimum allowable distance from the master link of the sling to the top of the canister is 66.1 inches. A minimum distance of 67 inches is specified in Sections 8.1.2, 8.2 and Appendix 12A, Section 4.5.4.

The structural adequacy of the canister structural lid and weld was evaluated using a finite element representation of the upper portion of the canister using the ANSYS program. As shown in Figure 3.4.3.2-1, the model represents one-half (180° section) of the upper 50-in. of the canister (including the structural and shield lids). The lids and shell in the model are comprised of SOLID45 elements. CONTACT52 elements are used to model the interaction between the structural lid and the canister shell and between the shield lid and the canister shell, just below the respective lid weld joints. The size of the CONTACT52 gaps was determined from nominal dimensions of contacting components. COMBIN40 elements are used between the structural and shield lids in the axial direction and between the shield lid and the backing ring. These gaps are assigned small gap sizes of 1E-8 inches. All gap/spring elements are assigned a stiffness of 1E8 lbs/in.

To enforce symmetry at the boundary of the model (in the x-y plane), all nodes on the x-y symmetry plane were restrained perpendicular to the symmetry plane (UZ). In addition, the nodes in the x-z plane at the bottom of the model were restrained in the axial direction.

Load-bearing members of a lifting device must be capable of lifting three (3) times the combined weight of the shipping container with which it will be used, plus the weight of intervening components of the lifting device without exceeding the tensile yield strength of their materials of construction. In addition, the lifting components must be capable of lifting five (5) times that combined weight without exceeding the ultimate tensile strength of the materials. NUREG 0612 also requires that the lifting loads must be based on the combined maximum static and dynamic loads that could be imparted on the handling device based on characteristics of the crane that will be used. A dynamic load factor of 10% has been applied.

To simulate the lifting of the canister by a three-point lifting device (the lifting configuration is actually two independent three-point lifting devices), point loads equal to one-third of the total canister and contents weight plus a dynamic loading factor of 10% were applied to the model. Because the model represents a half section of the canister, only two point loads were applied 120° apart as shown in Figure 3.4.3.2-1. Because of the symmetry conditions of the model, the force applied to the node on the symmetry plane was one-half of the value applied at the other location. A uniform temperature of 200°F is used in the model for modulus of elasticity. A temperature of 250°F is conservatively used to determine allowable stresses.

The maximum stress intensity generated in the canister model from the applied lifting forces was 3,753 psi, which occurs in the structural lid where the lifting loads were applied (see Figure 3.4.3.2-2).

As stated previously, the maximum stress intensity in the canister lifting model occurs in the structural lid that is constructed of Type 304L stainless steel. The yield strength (S_y) of Type 304L stainless steel at 250°F is 20,300 psi. The ultimate strength (S_u) of Type 304L stainless steel at 250°F is 63,550 psi.

The factor of safety (FS) for the canister lift based on yield strength is:

$$FS = \frac{S_y}{S_{INT}} = \frac{20{,}300\,\text{psi}}{3{,}753\,\text{psi}} = 5.40 > 3$$

The factor of safety for the canister lift based on ultimate strength is:

$$FS = \frac{S_u}{S_{INT}} = \frac{63{,}550\,\text{psi}}{3{,}753\,\text{psi}} = 16.93 > 5$$

Figure 3.4.3.2-1 Canister Lift Finite Element Model

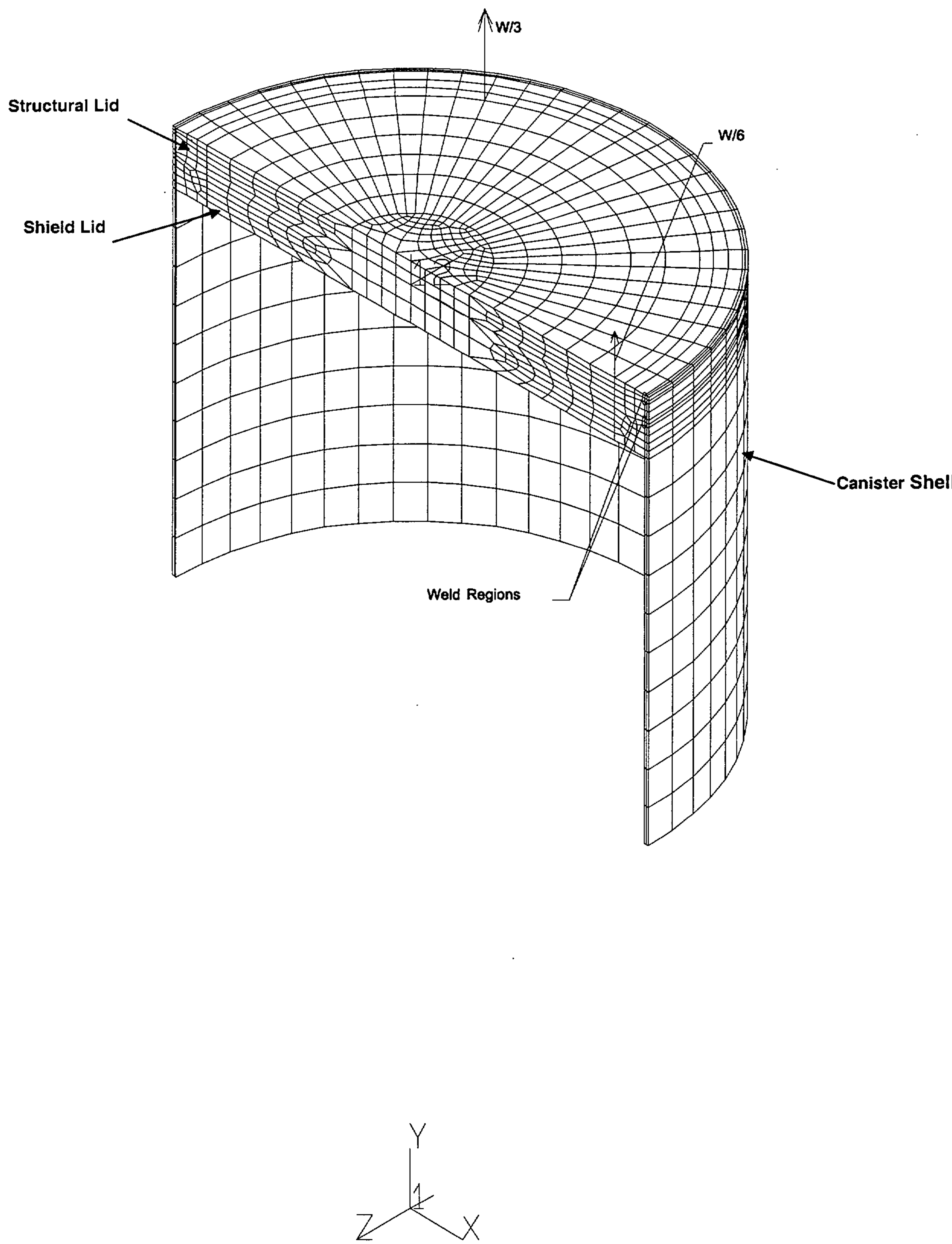
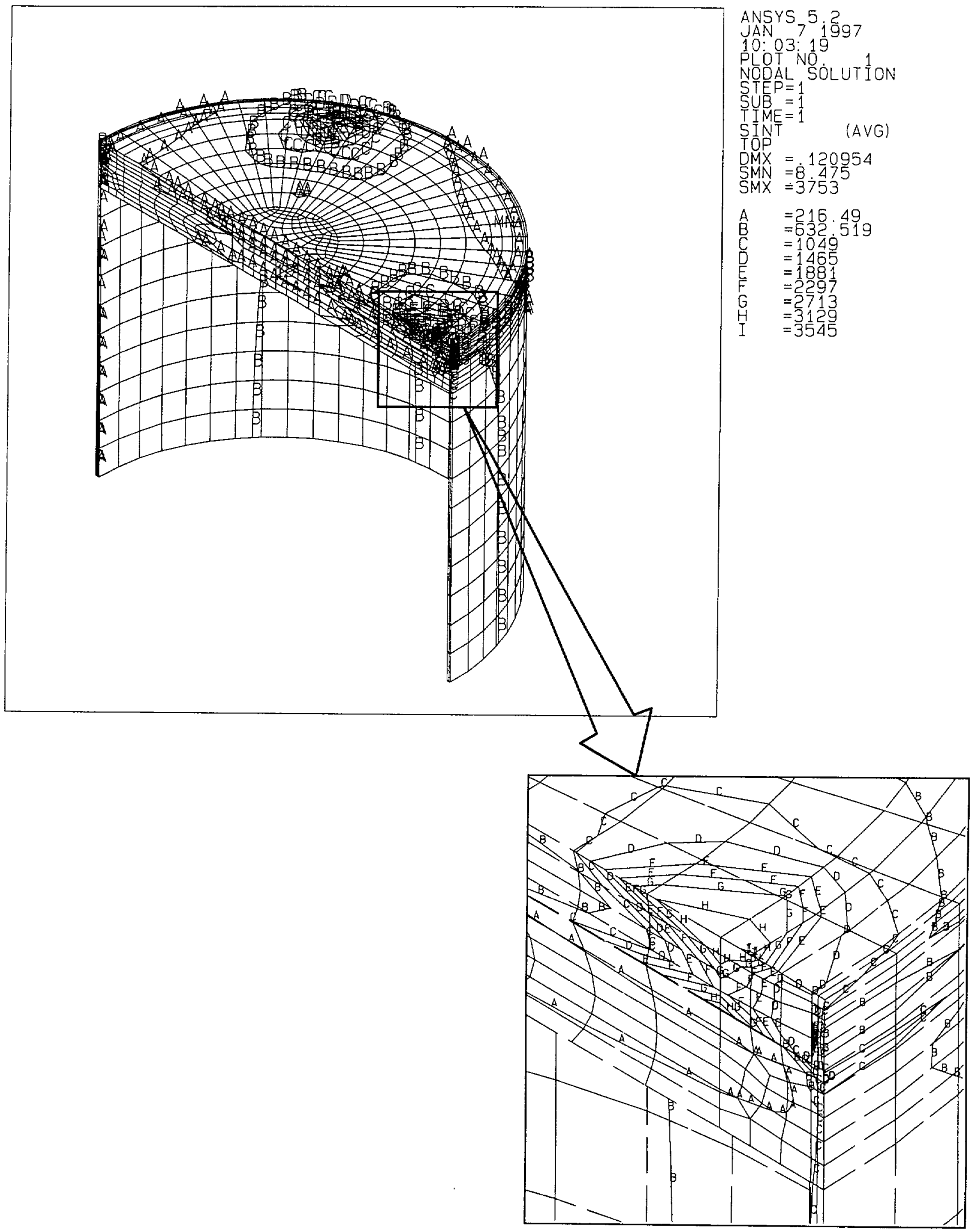


Figure 3.4.3.2-2 Canister Lift Model Stresses Intensity Contours (psi)



3.4.3.3 Transfer Cask Lift

The evaluation of the transfer cask presented here shows that the design meets ANSI N14.6 and NUREG 0612 for heavy lifts. The adequacy of the transfer cask is shown by evaluating the stress levels in all of the load-path components. The maximum weight of the loaded transfer cask is calculated to be 143,013 pounds (Table 3.2-1). For this analysis, the transfer cask is conservatively assumed to weigh 150,000 pounds. A dynamic load factor of 10 percent is applied to establish a design basis loaded canister weight of 165,000 pounds.

3.4.3.3.1 Transfer Cask Shell and Trunnion

A structural evaluation was prepared for the transfer cask to evaluate the structural adequacy of the cask shell and trunnion during lifting conditions, in accordance with ANSI N14.6 and NUREG 0612.

An ANSYS 3-D model is used to evaluate the lifting of a fully loaded transfer cask. Because of symmetry, the 3D ANSYS model considers one quarter of the transfer cask. The model contains only the upper portion of the transfer cask since the purpose of this calculation is to evaluate the shells at the trunnion region. The lead and NS-4-FR between the inner and outer shells of the transfer cask are also neglected since they are not structural components. SOLID95 and SHELL93 elements are used to model the trunnion and shells, respectively. BEAM4 elements are used at the interface of the trunnion and the inner and outer shells to transfer bending moment from the SOLID95 elements, which do not have a rotational degree of freedom, to the SHELL93 elements. The ANSYS model is shown in Figure 3.4.3.3-1.

The total (design) load for the transfer cask is conservatively assumed to be 165,000 pounds, including a 10% dynamic load factor. The loading applied to the model is (165,000) / 4 = 41,250 pounds. The load is applied upward at the trunnion as a "surface load." The lifting yoke dimensions determine the location of the load. The maximum temperature in the transfer cask shell/trunnion region is 300°F. A uniform temperature of 300°F is used in the model for the modulus of elasticity.

Per ANSI N14.6 and NUREG 0612, factors of safety of 6 on material yield strength and 10 on material ultimate strength are required. For the ASTM A-588 shell material, the yield strength, S_y, is 45.6 ksi, and the ultimate strength, S_u, is 70 ksi at 300°F. The trunnions are constructed of

ASTM A-350 carbon steel, Grade LF2. From the ASME Code, Section II, Part D, Tables U (S_u) and Y-1 (S_y), the yield stress, S_y is found to be 31.9 ksi, and the ultimate stress, S_u, is found to be 70 ksi at 300°F.

Tables 3.4.3.3-1 and 3.4.3.3-2 provide a summary of the top 30 maximum stresses for the outer shell and inner shell, respectively (see Figures 3.4.3.3-2 and 3.4.3.3-3 for node locations for outer shell and inner shell, respectively). Stress contour plots for the outer shell and inner shell are shown in Figures 3.4.3.3-4 and 3.4.3.3-5, respectively. As shown in Table 3.4.3.3-1 and 3.4.3.3-2, all stresses, except the local stresses, meet the requirement of ANSI N14.6 and NUREG 0612 with a factor of safety of 6 on material yield strength and 10 on material ultimate strength. Per ANSI N14.6, Section 4.2.1.2, the high local stresses, as categorized in accordance with ASME Code, Section III, NB-3213.10, are relieved by slight local material yielding and the stress design factors are not applicable.

The localized stresses occur at the interface of the trunnion with the inner and outer shells. The meridional direction lengths of the high stress areas are 2.8 inches and 2.0 inches for the inner shell and outer shell, respectively. In accordance with ASME NB-3213.10, the area of localized stresses is limited in the meridional direction to:

$$1.0\sqrt{Rt}$$

where:

R is the minimum midsurface radius, and
t is the minimum thickness in the region considered.

Based on this formula, the meridional direction length limitations for local stress regions are 5.2 inch (>2.8 inch) and 7.3 inch (>2.0 inch) for the inner and outer shells, respectively.

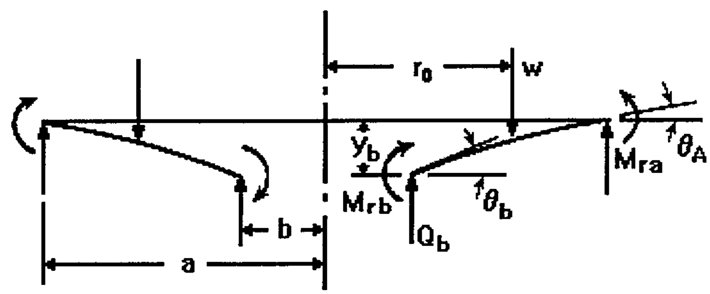
For the trunnion, the maximum bending and shear stress occur at the interface with the outer shell. The linearized stresses through the trunnion is 3,287 psi in bending and 1,221 psi in shear. Comparing to the material yield stress and ultimate stress (A350 carbon steel), the F.S. on yield and ultimate strength are 9.7 (> 6) and 21.3 (> 10), respectively.

3.4.3.3.2 Retaining Ring and Bolts

A retaining ring is bolted on the top of the transfer cask to prevent the inadvertent lifting of the canister out of the transfer cask, which could result in increased radiation exposure to nearby workers. In the event that the loaded transfer cask is inadvertently lifted during handling of the canister, the retaining ring and bolts must have sufficient strength to support the weight of the transfer cask. This event is evaluated in this section as an off-normal condition.

Retaining Ring

To qualify the retaining ring, the equations presented in Roark for annular rings are utilized. The retaining ring is represented as shown in the sketch below (Roark, Table 24, Case 1e). The following sketch assists in defining the variables used to calculate the stress in the retaining ring and bolts. The model assumes a uniform annular line load w applied at radius r_o.



The boundary conditions for the model are outer edge fixed, inner edge free.

The material properties and parameters for the analysis are:

Plate dimensions:

thickness:

$t = 0.75$ in

bolt circle:

$a = 39.2$ in

outer radius (outer edge)

$c = 40.4$ in

inner radius:

$b = 34.3$ in

Weight of transfer cask (with 10% dynamic load factor):

$wt = 81{,}000 \text{ lbs} \times 1.1$

Radial location of applied load: (canister outer radius)

$r_o = 35.3$ in

Ring material:

ASTM A588

Modulus of elasticity:

$E = 28.3 \times 10^6$ psi

Poisson's ratio:

$\nu = 0.31$

Number of bolts:

$Nb = 24$

Radial length of applied load:

$L_r = 2\pi r_o$

$L_r = 221.8$ in

Applied unit load:

$$w \equiv \frac{wt}{L_r}$$

$w = 401.7$ psi

The shear modulus is:

$$G = \frac{E}{2 \cdot (1 + v)}$$

$$= 1.08 \text{ x } 10^7 \text{ psi}$$

A plate constant, D, is used in determining boundary values and in the general equations for deflection, slope, moment and shear.

$$D = \frac{E \cdot t^3}{12 \cdot (1 - v^2)}$$

$$D = 1.101 \text{ x } 10^6 \text{ lbs-in}$$

Tangential shear constants, K_{sb} and K_{sro}, are used in determining the deflection due to shear:

$$K_{sro} = 1.2 \cdot \frac{r_o}{a} \cdot \ln\left(\frac{a}{r_o}\right)$$

$$= -0.113$$

$$K_{sb} = K_{sro}$$

The calculated shear stresses at points b and a (inner and outer radius) are:

$$\tau_b = 0 \text{ psi}$$

$$\tau_a = 482.3 \text{ psi}$$

The calculated radial bending stresses at points b and a (inner and outer radius) are:

$$\sigma_r(b) = 0 \text{ psi}$$

$$\sigma_r(a) = 15{,}220 \text{ psi}$$

The calculated tangential bending stresses at points b and a (inner and outer radius) are:

$$\sigma_t(b) = 828.0 \text{ psi}$$

$$\sigma_t(a) = 4{,}717.9 \text{ psi}$$

The principal stresses at the outer radius are:

$$\sigma_{1a} = \left(\frac{\sigma_r(a) + \sigma_t(a)}{2}\right) + \sqrt{\tau_a^{\,2} + \left(\frac{\sigma_r(a) - \sigma_t(a)}{2}\right)^2}$$

$\sigma_{1a} = 15{,}240$ psi

$$\sigma_{2a} = \left(\frac{\sigma_r(a) + \sigma_t(a)}{2}\right) - \sqrt{\tau_a^{\,2} + \left(\frac{\sigma_r(a) - \sigma_t(a)}{2}\right)^2}$$

$\sigma_{2a} = 4{,}695.8$ psi

$\sigma_{3a} = 0$ psi

The stress intensity at the outer radius ($P_m + P_b$) is:

$SI_a = \sigma_{1a} - \sigma_{3a}$

$SI_a = 15{,}240$ psi

The principal stresses at the inner radius are:

$$\sigma_{1b} = \left(\frac{\sigma_r(b) + \sigma_t(b)}{2}\right) + \sqrt{\tau_b^{\,2} + \left(\frac{\sigma_r(b) - \sigma_t(b)}{2}\right)^2}$$

$\sigma_{1b} = 0$ psi

$$\sigma_{2b} = \left(\frac{\sigma_r(b) + \sigma_t(b)}{2}\right) - \sqrt{\tau_b^{\,2} + \left(\frac{\sigma_r(b) - \sigma_t(b)}{2}\right)^2}$$

$\sigma_{2b} = -\ 828.0$ psi

$\sigma_{3b} = 0$ psi

The stress intensity at the inner radius ($P_m + P_b$) is:

$$SI_b = \sigma_{1b} - \sigma_{2b}$$

$$SI_b = 828.0 \text{ psi}$$

The maximum stress intensity occurs at the outer radius of the retaining ring. For off-normal conditions, the allowable stress is equal to the lesser of 1.8 S_m and 1.5 S_u (Service Level C, ASME Code, Subsection NB). For ASTM A588, the allowable stress at 300°F is 1.8 S_m = 1.8(23.3) = 41.94 ksi. The calculated stress intensity of 15.24 ksi is less than the allowable stress and the margin of safety is +1.75.

The bearing stress between the retaining ring and canister is calculated below:

Weight of Transfer Cask = 81,000 x 1.1 = 89,100 lbs

Area of contact between retaining ring and canister:

$$A = \pi\left(35.3^2 - 34.2^2\right) = 240 \text{ in}^2$$

$$S_{brg} = \frac{89{,}100}{240} = 371 \text{ psi}$$

Bearing stress allowable is S_y. For ASTM A588, the allowable stress at 300°F is 45.6 ksi. The calculated bearing stress is well below the allowable stress with a large margin of safety.

Shearing stress of Retaining Plate under the bolt head:

Outside diameter of bolt head = 1.125 in.

Total shearing area under the bolt head = $\pi(1.125)(24)(0.75)=63.6 \text{ in}^2$

Shearing stress under the bolt head = 89,100/63.6 = 1,401 psi

Allowable Stress = 0.60 x S_m = 0.60 x 23.3 = 13.98 ksi

Margin of Safety = (13,980/1401) – 1 = + large

Bolt Edge Distance:

Using Table J3.5 "Minimum Edge Distance, in." of Section J3 from "Manual of Steel Construction Allowable Stress Design," the required edge distance for a 0.75-inch bolt is 1.00 inch for a saw-cut edge. The actual edge distance for the bolts is:

$$\frac{80.8 - 78.31}{2} = 1.25\,\text{inch} \rangle 1.0\,\text{inch}$$

<u>Retaining Ring Bolts</u>

The load on a single bolt, F_F, due to the reactive force caused by inadvertently lifting the transfer cask, is:

$$F_F = \frac{wt}{Nb} \quad = 3{,}713 \text{ lbs}$$

The load on each bolt, F_M, due to the bending moment, is:

$$F_M = \left(\frac{2 \cdot \pi \cdot a}{Nb}\right) \cdot \left(\frac{\sigma \cdot t^2}{6 \cdot L}\right)$$

$$F_M = 12{,}200 \text{ lbs}$$

where:

σ = the radial bending stress at point a (15,220 psi)

L = the distance between the bolt center line and ring outer edge (c - a = 1.2 inches)

The total tension, F, on each bolt is:

$$F = F_F + F_M \; = 15{,}910 \text{ lbs}$$

The bolt cross-sectional area, A_b, is 0.4418 in^2. The bolt tensile stress is:

$$\sigma_t = \frac{F}{A_b} = 36{,}012 \text{ psi}$$

For off-normal conditions, the allowable stress for primary membrane stress in a bolt is $2S_m$. The allowable stress for SA193 Grade B6 high alloy steel bolts is 2 x 27.0 = 54.0 ksi at 200°F. The margin of safety for the bolts is +0.50.

The top plate internal threads are evaluated for resistance to shear pull out. The thread specifications are 3/4-10 UNC (Class 2A external thread and class 2B internal thread):

D = 0.7482 in. (basic major diameter of the bolt threads)
n = 10 (number of bolt threads per inch)
$D_s min$ = 0.7353 in.(minimum major diameter of bolt threads)
$E_n max$ = 0.6927 in. (maximum pitch diameter of lid threads)
L_e = 2.25 - 0.75 = 1.5 in. (minimum thread engagement)

For the top plate (ASTM 588) at a temperature of 200°F, the yield and ultimate stresses are:

$S_y = 47.5$ ksi
$S_u = 70.0$ ksi
$S_m = 23.3$ ksi

The shear area for the internal (top plate) threads (A_n) is:

$$A_n = 3.1416 n L_e D_s min \left[\frac{1}{2n} + 0.57735(D_s\,min - E_n\,max)\right] = 2.584\ in^2$$

The shear stress (τ_n) in the top plate is:

$$\tau_n = \frac{F_y}{A_n} = \frac{15{,}910}{2.584} = 6{,}157\ psi$$

where:

$$F_y = F = F_F + F_M$$

Conservatively, the shear allowable for normal conditions is used.

$$\tau_{allowable} = (S_m)\ (0.6) = (23.3\ ksi)(0.6) = 13.98\ ksi$$

The Margin of Safety is: (13,980/6,157) -1 = +1.27

3.4.3.3.3 Rails and Welds – Shield Door

This section demonstrates the adequacy of the shield doors, door rails, and welds in accordance with ANSI N14.6, which requires safety factors of 6 and 10 on material yield strength and ultimate strength, respectively.

The shield door rails support the weight of a wet, fully loaded canister and the weight of the shielding doors themselves. The shield doors are 9.5-inch thick plates resting on top of the bottom section of the door rails. The door rails are 9.88 inches deep by 6.5 inches thick and are welded to the bottom plate of the transfer cask. Both the doors and the rails are constructed of A-350, Grade LF2 carbon steel. The design load for the rails (considering 10% dynamic factor) is conservatively assumed to be:

$$W = 150{,}000 \times 1.1 = 165{,}000 \text{ lbs}$$

This evaluation shows that the shield doors, door rail structures, and welds are adequate to support a wet, fully loaded canister.

Operating Conditions

Based on the thermal analysis as presented in Chapter 4, the maximum calculated temperature for the transfer cask doors is 257°F. The material allowables are conservatively taken at 300°F.

Material Properties

The material properties for the ASTM A-350, Grade LF2, carbon steel are taken from the ASME Code, Division II, Part D, Tables U (S_u) and Y-1 (S_y).

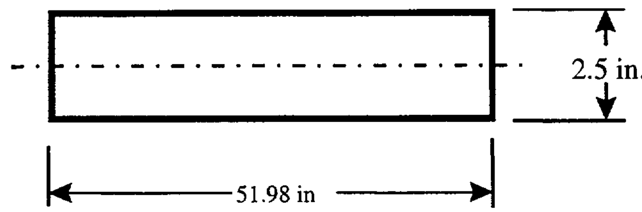
Yield Strength (S_y) = 31.9 ksi at 300° F (ASTM A-350)
Ultimate Strength (S_u)= 70.0 ksi at 300°F (ASTM A-350)

Stress Evaluation for Door Rail

The shear stress in each door rail bottom plate due to the applied load of W is:

$$\tau = \frac{W}{2 \times A_S} = \frac{165}{2 \times 129.95} = 0.64 \text{ ksi}$$

where:



A_s = Shear area

$= L \times t = 129.95 \text{ in}^2$,

and,

L = rail length supporting doors = 2 x (45.38-17.39- 4/2) = 51.98 inch,

t = bottom plate thickness = 2.5 inch.

The bending stress in each rail bottom section due to the applied load of W is:

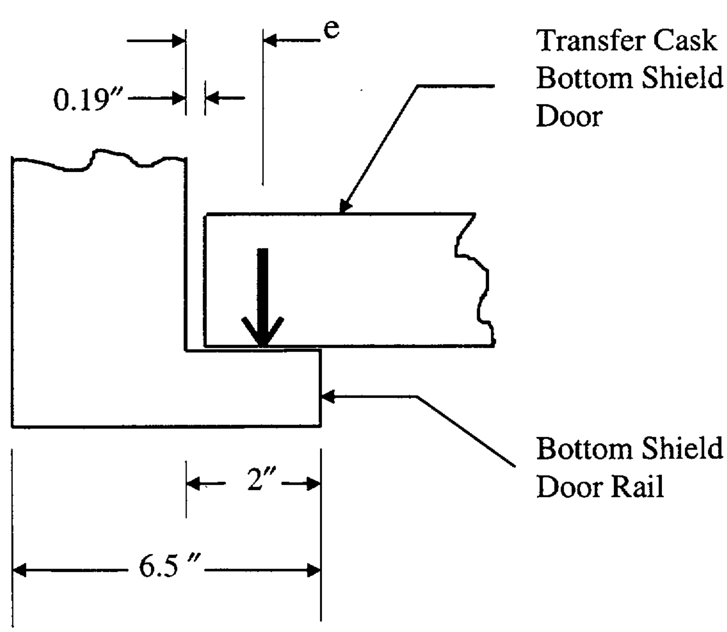
$$\sigma_b = \frac{M}{S} = \frac{90.8}{54.15} = 1.68 \text{ ksi},$$

where:

M = moment at the bottom section,

$$= \frac{W}{2} \text{ x e} = \frac{165}{2} \times 1.1$$

= 90.8 in-kips,



and,

$$e = \frac{2-0.19}{2} + 0.19 = 1.1 \text{ inch applied load moment arm.}$$

$$S = \frac{b \times d^2}{6} = \frac{51.98 \times 2.5^2}{6}$$

$$= 54.15 \ \text{in}^3.$$

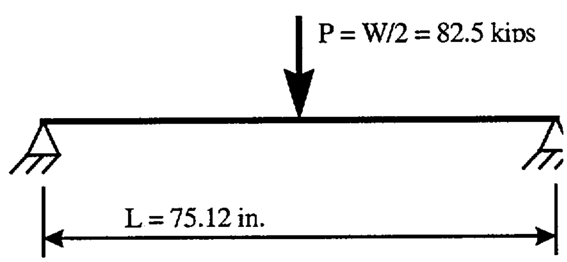
Per ANSI N14.6, Section 4.2.1, shear stress or maximum tensile stress are to be compared with material yield and ultimate strength. The resulting factors of safety are:

$$\varphi_y = \frac{31.9}{1.68} = 19.0 > 6 \qquad \text{(For yield strength criteria)}$$

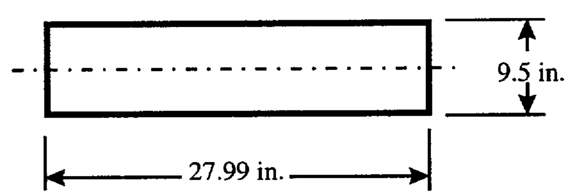
$$\varphi_u = \frac{70}{1.68} = 41.7 > 10 \qquad \text{(For ultimate strength criteria)}$$

<u>Stress Evaluation for the Shield Doors</u>

The shield doors consist of two 9.5-inch thick plates that rest on top of the rails. Stresses can be approximated by modeling the shield door as a simply supported beam with a concentrated load at the center.



The shear stress in each shield door is:



(Shield Door Cross Section)

$$\tau = \frac{W}{2 \times A_S} = \frac{165}{2 \times (27.99 \times 9.5)}$$

$$= 0.31 \text{ ksi}$$

and the bending stress in each shield door is

$$\sigma_b = \frac{M}{S} = \frac{1549.35}{421} = 3.68 \text{ ksi},$$

where:

M = moment at the bottom section,

$$= \frac{P \times L}{4} = \frac{82.5 \times 75.12}{4} = 1549.35 \text{ in-kips},$$

S = section modulus

$$= \frac{b \times d^2}{6} = \frac{27.99 \times 9.5^2}{6} = 421 \text{ in}^3$$

The factors of safety are:

$$\varphi_y = \frac{31.9}{3.68} = 8.7 > 6 \qquad \text{(For yield strength criteria)}$$

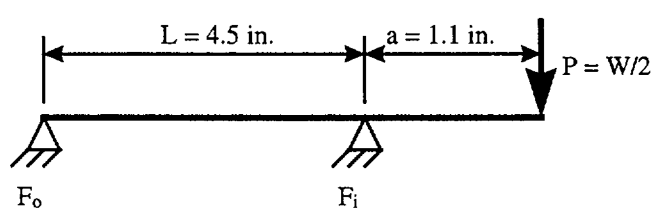
$$\varphi_u = \frac{70}{3.68} = 19.0 > 10 \qquad \text{(For ultimate strength criteria)}$$

Door Rail Weld Evaluation

The rail welds were evaluated by determining the reactive forces, F_o and F_i, experienced by the outer and inner welds due to applied load, W.

$$F_o = \frac{P \times a}{L} = \frac{165}{2} \times \frac{1.1}{4.5}$$

$$= 20.17 \text{ kips}$$



$$F_i = \frac{165}{2} + 20.17 = 102.67 \text{ kips}$$

Maximum stresses at the groove weld (size = 0.75 inch) are:

$$= \frac{102.67}{56.25 \times 0.75} = 2.4 \text{ ksi}$$

The factors of safety are:

$$\varphi_y = \frac{31.9}{2.4} = 13.3 > 6$$ (For yield strength criteria)

$$\varphi_u = \frac{70}{2.4} = 29.2 > 10$$ (For ultimate strength criteria)

Figure 3.4.3.3-1 Finite Element Model for Transfer Cask Trunnion and Shells

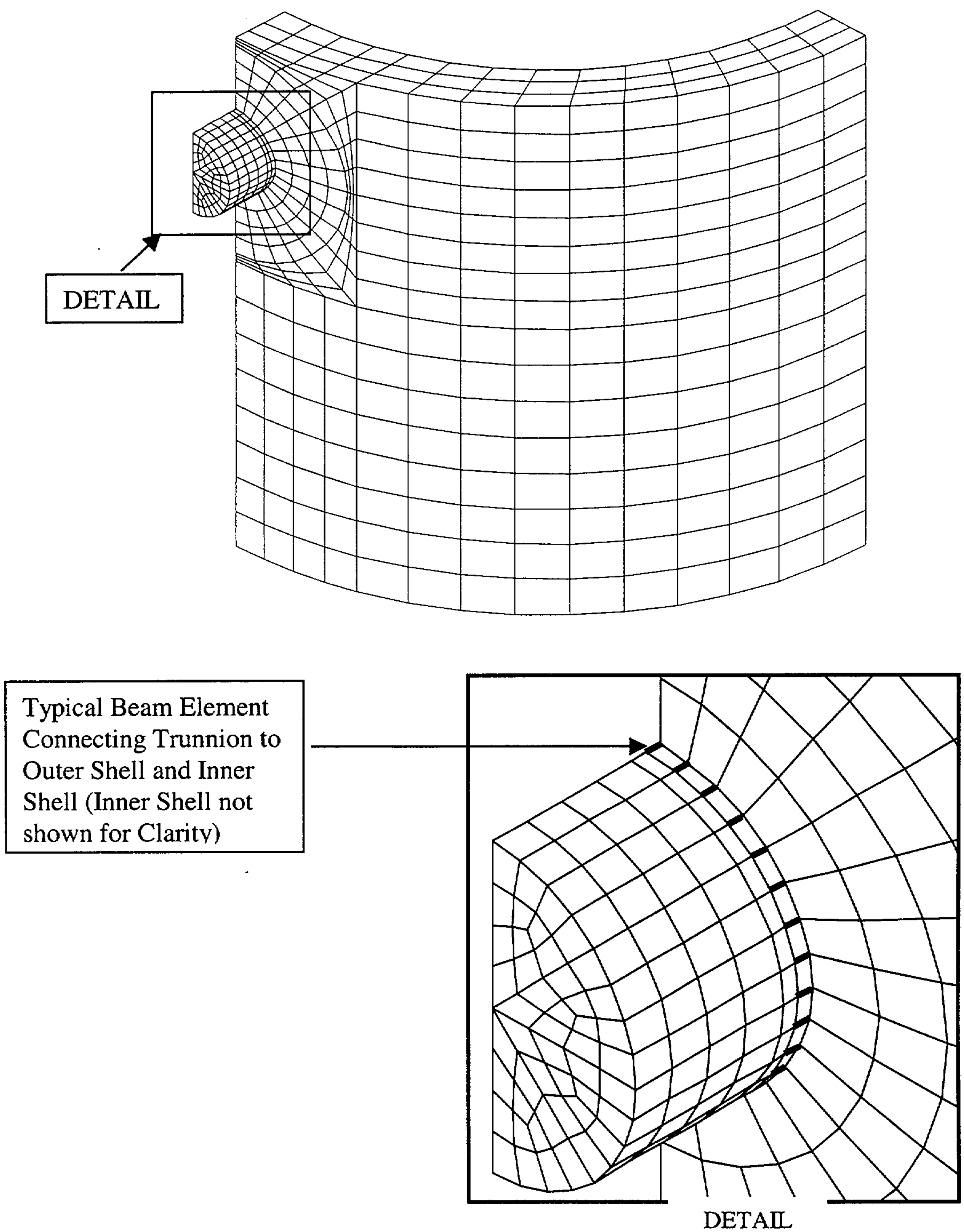


Figure 3.4.3.3-2 Node Locations for Transfer Cask Outer Shell Adjacent To Trunnion

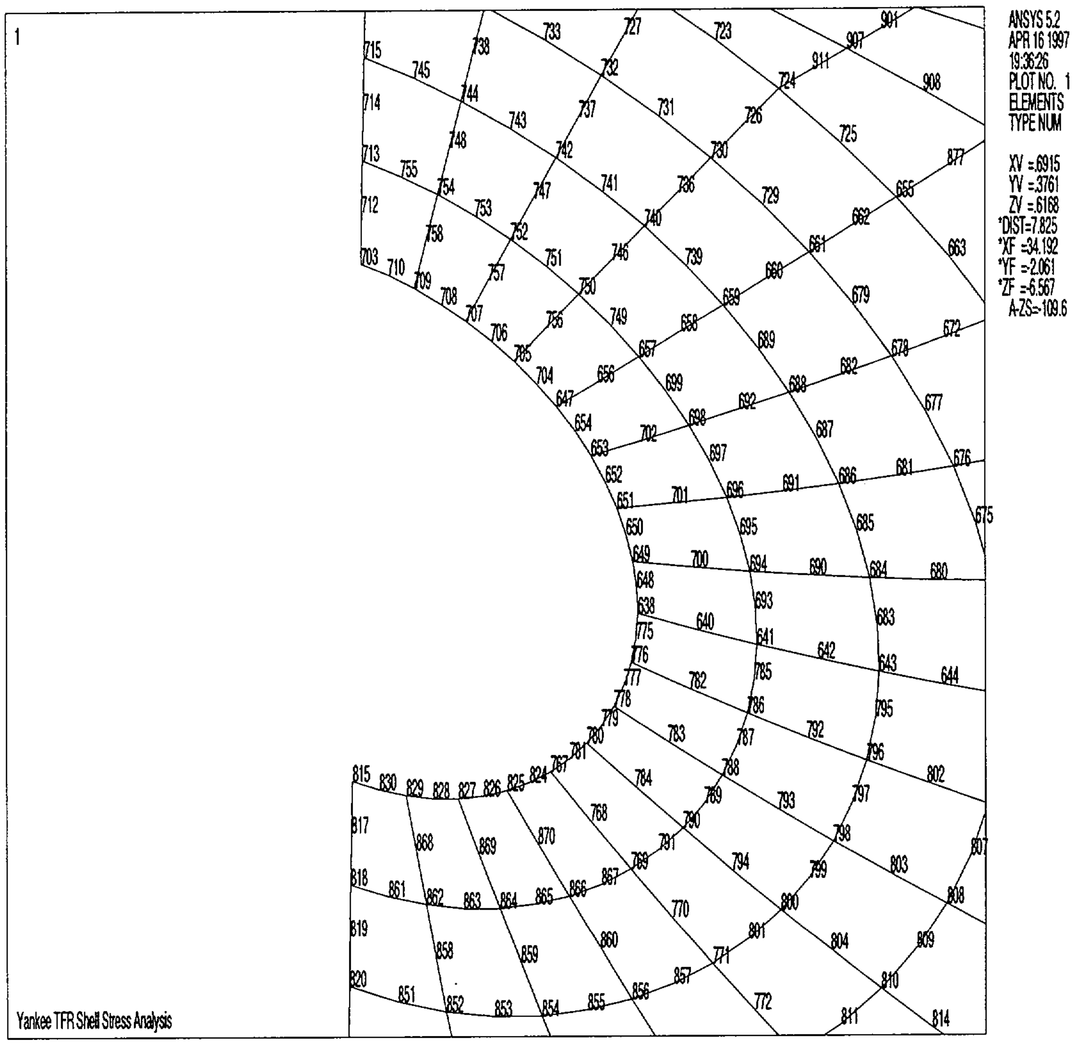


Figure 3.4.3.3-3 Node Locations for Transfer Cask Inner Shell Adjacent To Trunnion

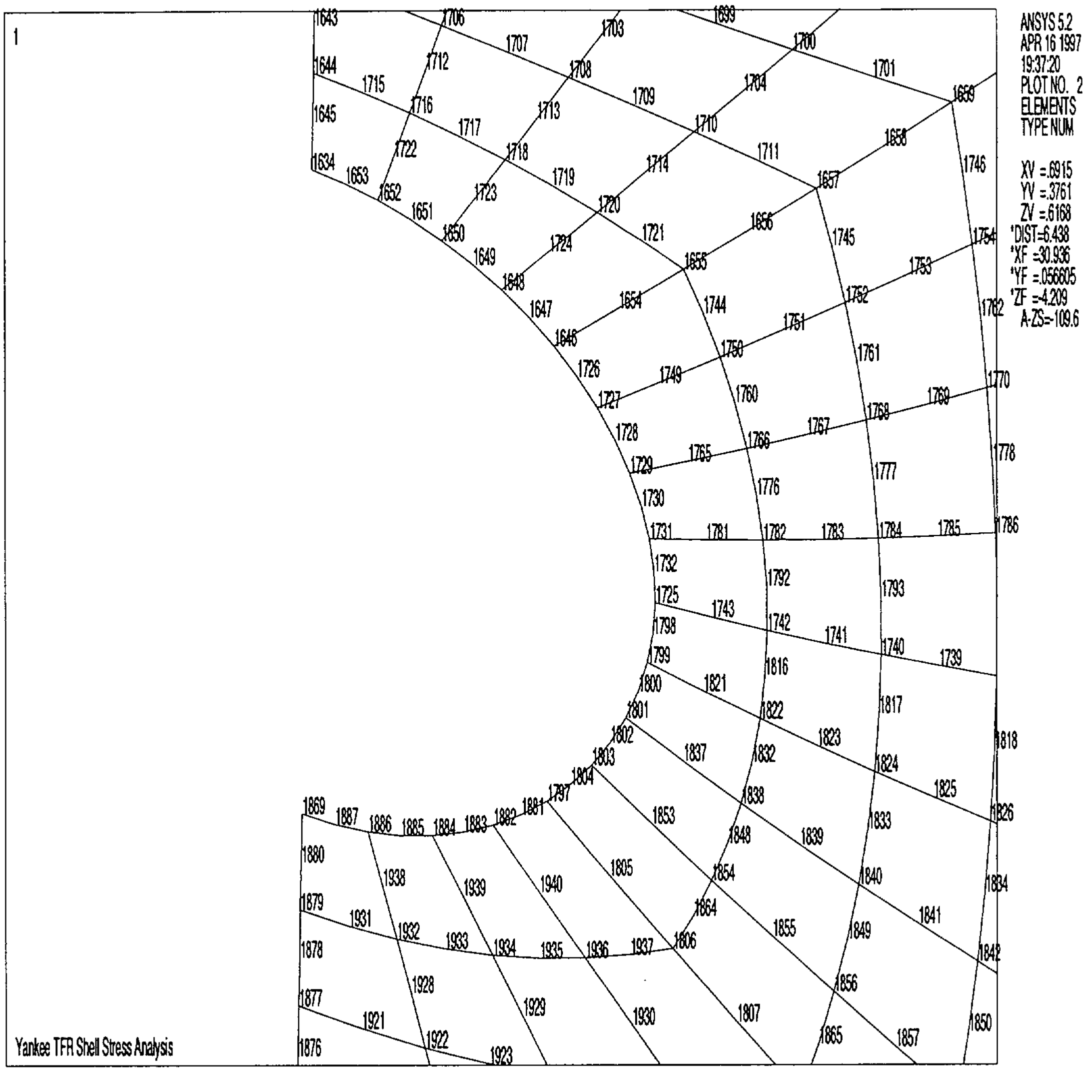


Figure 3.4.3.3-4 Stress Contours for Transfer Cask Outer Shell

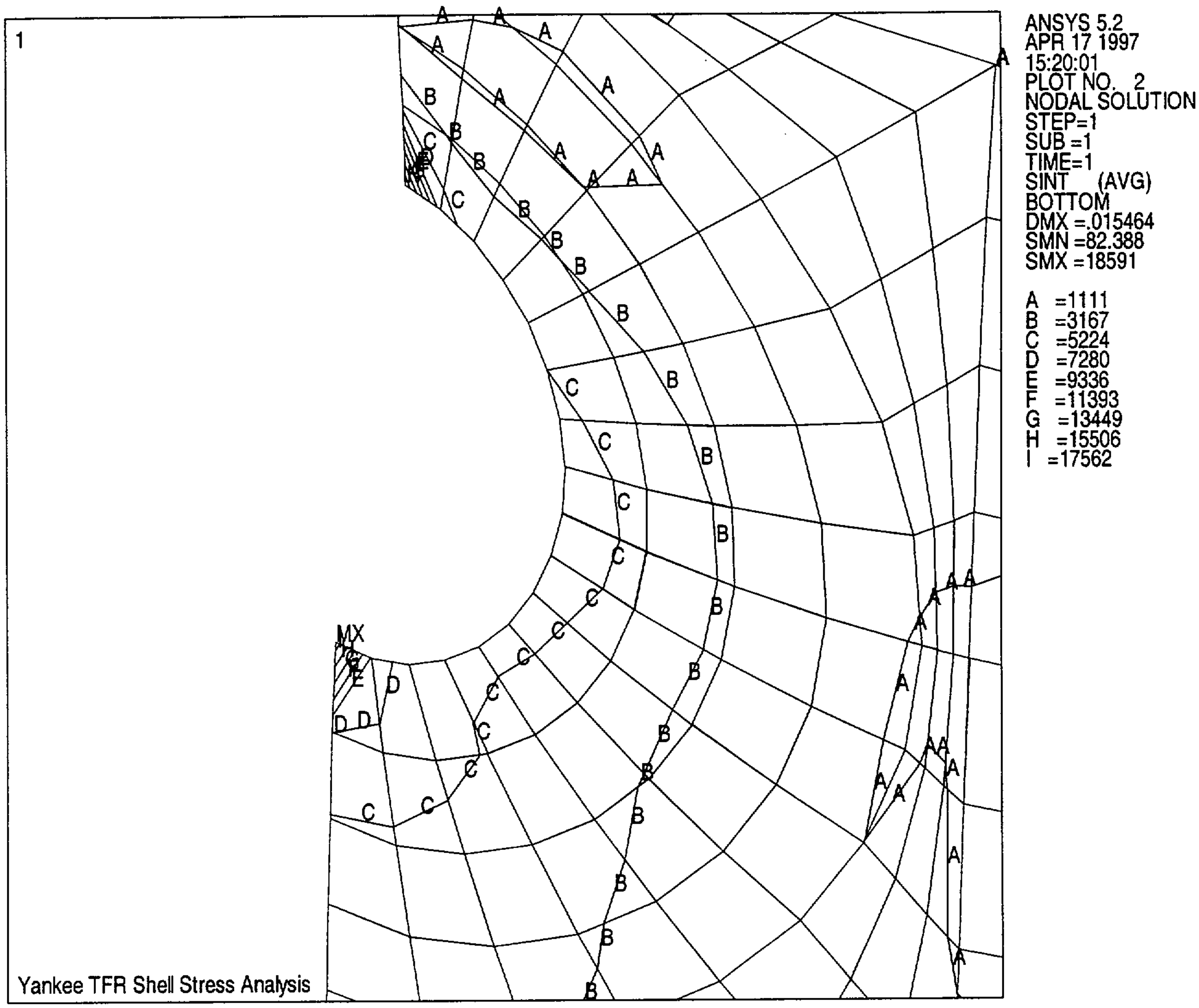


Figure 3.4.3.3-5 Stress Contours for Transfer Cask Inner Shell

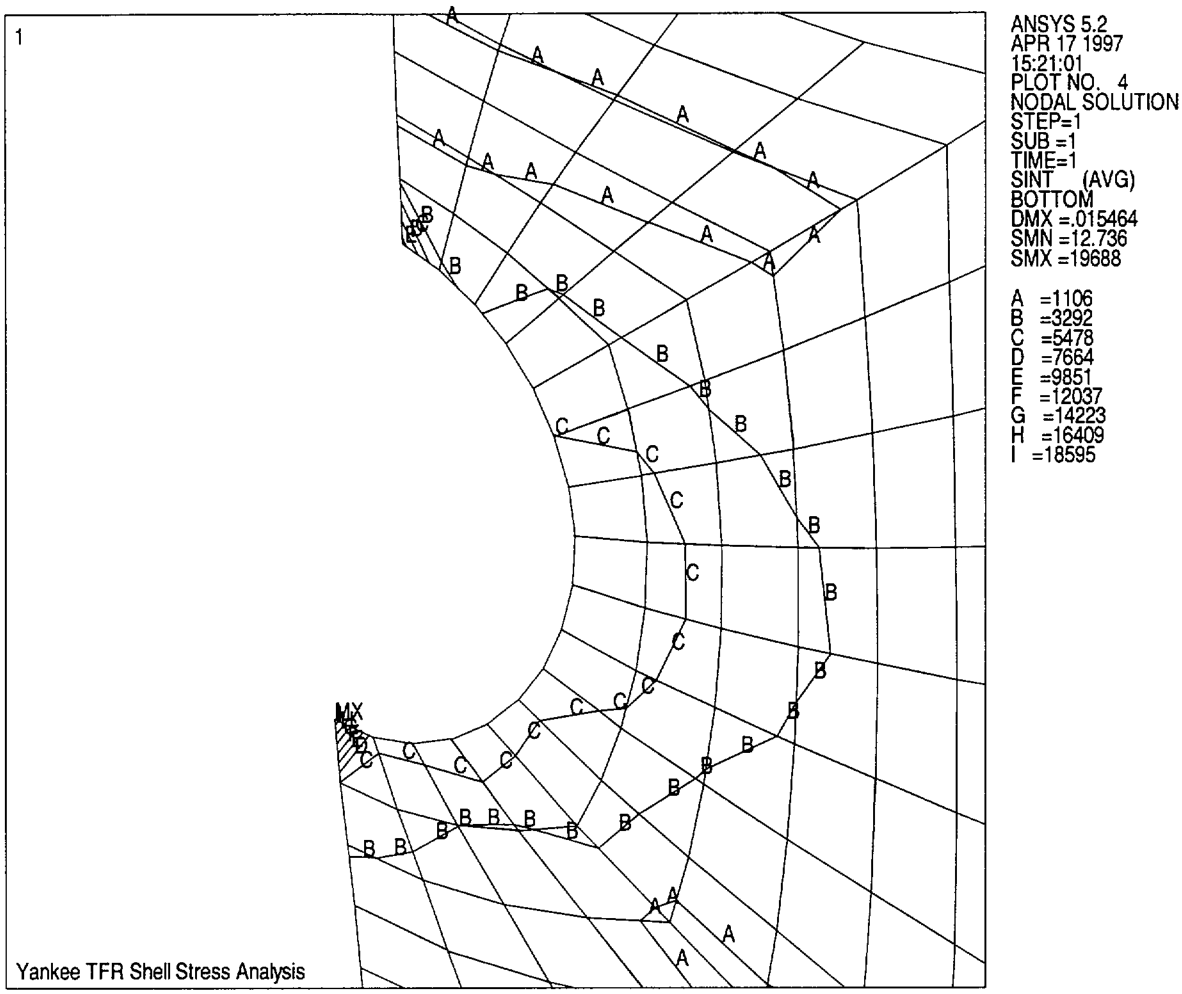


Table 3.4.3.3-1 Top 30 Stresses for Transfer Cask Outer Shell

Node[1]	Principal Stresses (psi) S1	S2	S3	Nodal S.I. (psi)	F.S. on Yield[2] (S_y/S.I.)	F.S. on Ultimate[2] (S_u/S.I.)
815	18395	1593.7	-195.14	18591[3]	N/A	N/A
703	512.98	-1170.3	-16452	16965[3]	N/A	N/A
829	8895.7	4797.9	-20.588	8916.2[3]	N/A	N/A
818	7296.6	1839.7	-7.8552	7304.4[3]	N/A	N/A
862	6536.6	2745.7	-9.4784	6546	7.0	10.7
638	3595.9	-19.633	-2848.4	6444.3	7.1	10.9
776	4756.4	314.84	-1459.5	6215.9	7.3	11.3
871	18.466	-866.71	-6128.8	6147.2	7.4	11.4
709	93.408	-5325.4	-6033.2	6126.6	7.4	11.4
649	2094.5	-385.96	-3988.5	6083	7.5	11.5
827	5931.4	3499.1	-24.444	5955.9	7.7	11.8
778	5262.6	1098.1	-595.23	5857.9	7.8	11.9
864	5725.4	2444.8	-5.0196	5730.4	8.0	12.2
873	17.86	-668.51	-5669.4	5687.3	8.0	12.3
780	5345.3	2320.7	-313.59	5658.9	8.1	12.4
651	1052.3	-1275.6	-4516.8	5569.1	8.2	12.6
767	5246.3	3215.3	-188.33	5434.6	8.4	12.9
825	5317.3	3368.9	-91.577	5408.9	8.4	12.9
875	17.578	-482.15	-5250.6	5268.2	8.7	13.3
883	29.511	-789.43	-5198.3	5227.8	8.7	13.4
653	670.82	-2716.1	-4540.7	5211.5	8.7	13.4
866	5135.7	1721.4	0.66016	5135	8.9	13.6
820	5118.7	2375.9	-1.6525	5120.3	8.9	13.7
893	30.529	-567.18	-4905	4935.5	9.2	14.2
852	4898.5	2243.8	-2.5875	4901.1	9.3	14.3
769	4816.4	1032.1	-1.2375	4817.7	9.5	14.5
647	505.55	-3770.9	-4172.3	4677.9	9.7	15.0
641	2895.5	1.2697	-1776.8	4672.3	9.8	15.0
694	2169.8	4.4459	-2469.6	4639.4	9.8	15.1
903	32.924	-362.76	-4599.4	4632.4	9.8	15.1

Notes:
1. See Figure 3.4.3.3-2 for node locations.
2. S_y = 45,600 psi, S_u = 70,000 psi.
3. Local stresses that are relieved by local material yielding. Therefore, stress design factors of 6 and 10 on material yield and ultimate strength are not applicable (ANSI N14.6, Section 4.2.1.2).

Table 3.4.3.3-2 Top 30 Stresses for Transfer Cask Inner Shell

Node[1]	Principal Stresses (psi) S1	S2	S3	Nodal S.I. (psi)	F.S. on Yield[2] (S_y/S.I.)	F.S. on Ultimate[2] (S_u/S.I.)
1869	17689	436.58	-1999.3	19688[3]	N/A	N/A
1634	9754.8	619.16	-759.58	10514[3]	N/A	N/A
1882	7096.9	803.79	-559.18	7656[3]	N/A	N/A
1731	2629	-50.822	-4142.7	6771.7	6.7	10.3
1725	1528.8	-289.56	-5044.6	6573.4	6.9	10.6
1884	5878.7	463.76	-401.23	6279.9	7.3	11.1
1729	3436.2	79.26	-2823.1	6259.4	7.3	11.2
1742	2276.9	0.49938	-3877.4	6154.4	7.4	11.4
1782	2681.2	0.15472	-3465.3	6146.5	7.4	11.4
1797	936.56	190.95	-5161.7	6098.2	7.5	11.5
1801	499.98	-2533.2	-5548.2	6048.1	7.5	11.6
1799	674.7	-1179.9	-5362.4	6037.1	7.6	11.6
1886	5937.1	2421.9	-85.981	6023.1	7.6	11.6
1803	469.31	-3500.1	-5469.9	5939.2	7.7	11.8
1822	1911.9	5.7018	-4006.7	5918.5	7.7	11.8
1766	2791.3	-1.7689	-2991.6	5783	7.9	12.1
1727	3960.6	331.95	-1535.6	5496.2	8.3	12.7
1879	5014.2	118.74	-303.37	5317.6	8.6	13.2
1838	1675.8	22.011	-3579.3	5255.1	8.7	13.3
1646	4023.3	708.54	-1131	5154.4	8.8	13.6
1750	2409.6	-7.2699	-2506.8	4916.4	9.3	14.2
1740	2413.9	-1.0861	-2489.3	4903.3	9.3	14.3
1784	2271.4	-0.2874	-2557.6	4829	9.4	14.5
1824	2217	-0.4888	-1981.1	4198.1	10.9	16.7
1854	1463.1	39.789	-2681.4	4144.5	11.0	16.9
1806	3183.2	68.412	-943.43	4126.6	11.1	17.0
1768	1787.3	-0.4604	-2229.5	4016.7	11.4	17.4
1648	3076.7	1254.1	-849.56	3926.2	11.6	17.8
1932	3633.4	1174.9	-9.7208	3643.1	12.5	19.2
1738	2162.3	1.0241	-1473.5	3635.8	12.5	19.3

Notes:
1. See Figure 3.4.3.3-3 for node locations.
2. S_y = 45,600 psi, S_u = 70,000 psi.
3. Local stresses that are relieved by local material yielding. Therefore, stress design factors of 6 and 10 on material yield and ultimate strength are not applicable (ANSI N14.6, Section 4.2.1.2).

3.4.4 NAC-MPC Components Under Normal Operating Loads

The NAC-MPC system is evaluated using individual finite element models for the fuel basket, canister, and vertical concrete cask. Since the individual components are free to expand without interference, the structural finite element models need not be connected. Temperature dependent material properties are used in the finite element models based on calculated temperatures from the thermal analysis in Chapter 4.

3.4.4.1 Canister and Basket Analyses

3.4.4.1.1 Canister Thermal Stress Analysis

A three-dimensional finite element model of the canister was constructed using ANSYS SOLID45 elements. By taking advantage of the symmetry of the canister, the model represents one-half (180° section) of the canister, including the canister shell, bottom plate, structural lid, and shield lid. The model uses gap/spring elements to simulate contact between adjacent components. Specifically, contact between the structural and shield lids was modeled using COMBIN40 combination elements in the axial (UY) degree of freedom. Simulation of the backing ring is accomplished using a ring of COMBIN40 gap/spring elements connecting the shield lid and the canister in the axial direction at the lid lower outside radius. In addition, CONTAC52 elements were used to model the interaction between the structural lid and the canister shell and between the shield lid and canister shell, just below the respective lid weld joints. The size of the CONTAC52 gaps was determined from nominal dimensions of contacting components. The COMBIN40 elements used between the structural and shield lids and for the backing ring were assigned small gap sizes of 1E-8 inches. All gap/spring elements were assigned a stiffness of 1E+8 lbs/in. The three-dimensional ANSYS model of the canister used in the thermal stress evaluation is shown in Figure 3.4.4.1-1 through Figure 3.4.4.1-3.

The ANSYS thermal stress analysis was performed with canister temperatures that enveloped the canister temperature gradients for normal storage (100°F and -40°F ambient temperatures) and transfer conditions. Prior to performing the thermal stress analysis, the steady-state temperature distribution was determined using temperature information from the storage and transfer thermal analyses (Chapter 4). This was accomplished by converting the SOLID45 structural elements of the canister model to SOLID70 thermal elements and using the material properties from the thermal analyses. Nodal temperatures were applied at six key locations (i.e., top-center of the structural lid, top-outer diameter of the structural lid, bottom-center of the shield lid, bottom-center of the bottom plate, bottom-outer diameter of the bottom plate, and mid-elevation of the canister shell). The temperatures of the key locations used in the analysis were as follows:

Top center of the structural lid	=	140°F
Top outer diameter of the structural lid	=	100°F
Bottom center of the shield lid	=	165°F
Bottom center of the bottom plate	=	250°F
Bottom outer diameter of the bottom plate	=	130°F
Mid-elevation of the canister shell	=	500°F

The temperatures for all nodes in the canister model were obtained by the solution of the steady state thermal conduction problem.

These temperatures were selected to envelope the temperature differences experienced by the canister for storage and transfer conditions as calculated in the thermal analysis presented in Chapter 4. The following table shows the temperature differences (ΔT) of the canister in the radial and axial directions for the storage and transfer conditions and those used in the canister thermal stress analysis:

Condition	**Maximum ΔT (°F)**		
	Top of Structural Lid (Radial)	**Bottom Plate (Radial)**	**Canister Shell (Axial)**
Storage, Normal 75°F ambient	17	113	223
Storage, Off-Normal 100°F ambient	17	115	225
Storage, Off-Normal, -40°F ambient	15	104	213
Storage, Off-Normal Half Inlets Blocked	16	115	221
Transfer, 75°F ambient	5	21	386
Canister Thermal Stress Analysis	40	120	400

Additionally, canister temperatures used for determining allowable stress values were selected to envelope the maximum temperatures experienced by the canister during storage and transfer conditions. Specifically, allowable stresses were selected at temperatures of 250°F, 550°F, and 250°F for the structural/shield lid region, the canister shell, and the bottom plate region, respectively.

After solving for the canister temperature distribution, the thermal stress analysis was then performed by converting the SOLID70 elements back to SOLID45 structural elements. A single node at the centerline of the bottom plate is restrained in the axial direction (UY) to eliminate rigid body translation in the Y-direction. The nodes along the centerline of the structural and shield lids and the bottom plate were restrained in the x-direction (UX) to prevent rigid body motion in the x-direction. The nodes on the symmetry boundary face were restrained in the direction normal to the symmetry plane (UZ). A linear solution was performed to obtain the stresses due to thermal expansion.

The resulting maximum (secondary) thermal stresses in the canister are summarized in Table 3.4.4.1-1. The sectional stresses at 15 axial locations were obtained for each angular division of the model (a total of 21 angular locations for each axial location). The locations for the stress sections are shown in Figure 3.4.4.1-4.

3.4.4.1.2 Canister Dead Weight Load Analysis

The canister was structurally analyzed for dead weight load using the ANSYS model described in Section 3.4.4.1.1. The canister temperature distribution discussed in Section 3.4.4.1.1 was used in the dead load structural analysis to evaluate the material allowable stresses at temperature. The fuel and fuel basket assembly contained within the canister were not explicitly modeled, but were included in the analysis by applying a uniform pressure load representing their combined weight to the top surface of the canister bottom plate. The nodes on the bottom surface of the bottom plate were restrained in the axial direction in conjunction with the constraints described in Section 3.4.4.1.1. An acceleration of 1g was applied to the model in the axial direction (Y) to simulate the dead load.

The resulting maximum canister dead load stresses are summarized in Tables 3.4.4.1-2 and 3.4.4.1-3 for primary membrane and primary membrane plus bending stresses, respectively. The sectional stresses at 15 axial locations were obtained for each angular division of the model (a total of 21 angular locations for each axial location). The locations for the stress sections are shown in Figure 3.4.4.1-4.

The lid support ring is evaluated for the dead load condition using classical methods. The lid support ring is welded to the inner surface of the canister shell, under the shield lid. The lid support ring is made of ASTM A-479, Type 304 stainless steel. A temperature of 600°F is conservatively used to determine the material allowable stress. The total weight, W, imposed on the lid support ring is conservatively considered to be the weight of the structural lid (3,234 lbs), the shield lid (5,389 lbs) and the weight of the backing ring (16 lbs). The stresses on the support ring are the bearing stresses and shear stresses at its weld to the canister shell.

The bearing stress $\sigma_{bearing}$ is calculated as follows:

$$\sigma_{bearing} = \frac{W}{area}$$

$$= \frac{8639}{108}$$

$$= 80 \text{ psi}$$

where:

$W = 3,234 + 5,389 + 16$

$= 8,639$ lbs

area $= \pi \times D \times t$ $= 108\ in^2$

D = lid support ring average diameter = 68.89 inch

t = radial thickness of support ring = 0.5 inch

The yield strength (S_y) for A-479, Type 304 stainless steel is 18,600 psi @ 600°F. The allowable bearing stress is 1.0 S_y per ASME Section III, Subsection NB.

Margin of Safety = (18,600/80) -1

= + Large

The weld for the lid support ring is a 3/8 inch partial penetration groove weld. The total shear force on the weld is considered to be the weight of the structural and shield lids, the backing ring, and the lid support ring (8,655 lbs). The shear stress on the weld is calculated as follows:

$$\sigma_w = \frac{W}{\text{area}}$$

$$= \frac{8655}{81.32}$$

$$= 106 \text{ psi}$$

where:

area	= π x D x t_w	=	81.32 in^2
D	= shield lid diameter	=	69.03 inch
t_w	= weld size	=	0.375 inch

The yield strength (S_y) for A-479, Type 304 stainless steel is 18,600 psi @ 600°F. In accordance with ASME Section III, Subsection NB, the allowable shear stress is 0.6 x S_m.

Margin of Safety = (0.6 x (2/3) x 18,600) / 106 -1

= + Large

3.4.4.1.3 Canister Maximum Internal Pressure Analysis

The canister was structurally analyzed for a design internal pressure load of 11.5 psig using the ANSYS model and temperature distribution and restraints described in Section 3.4.4.1.1. This pressure bounds the calculated pressure of 7.9 psig for normal conditions. The design internal pressure of 11.5 psi is applied as a surface force to the elements along the internal surface of the canister shell, bottom plate, and shield lid.

The resulting maximum canister stresses for maximum internal pressure load are summarized in Tables 3.4.4.1-4 and 3.4.4.1-5 for primary membrane and primary membrane plus primary bending stresses, respectively. The sectional stresses at 15 axial locations were obtained for each angular division of the model (a total of 21 angular locations for each axial location). The locations of the stress sections are shown in Figure 3.4.4.1-4.

3.4.4.1.4 Canister Handling Analysis

The canister was structurally analyzed for handling loads using the ANSYS model and conditions described in Section 3.4.4.1.1. Normal handling of the canister was simulated by restraining the model at three lift points and applying a 1.1g acceleration load to the model in the axial direction, which includes a 10% dynamic load factor. The canister is lifted at six points; however, the handling analysis considers only a three-point lifting configuration. Since the model represents a one-half section of the canister, the three-point lift was simulated by restraining two nodes 120° apart (one node at the symmetry plane and a second node 120° from the first) along the bolt diameter at the top of the structural lid in the axial direction. Additionally, the nodes along the centerline of the lids and bottom plate were restrained in the radial direction, and the nodes along the symmetry face were restrained in the direction normal to the symmetry plane.

The resulting maximum stresses in the canister for the handling load are summarized in Tables 3.4.4.1-6 and 3.4.4.1-7 for primary membrane and primary membrane plus primary bending stresses, respectively. The sectional stresses at 15 axial locations were obtained for each angular division of the model (a total of 21 angular locations for each axial location). The locations for the stress sections are shown in Figure 3.4.4.1-4.

3.4.4.1.5 Canister Load Combination

The canister was structurally analyzed for the combined thermal, dead, maximum internal pressure, and handling loads using the ANSYS model and conditions described in Section 3.4.4.1.1. Loads were applied to the model as discussed in Sections 3.4.4.1.1 through 3.4.4.1.4. A maximum internal pressure of 11.5 psi was used in conjunction with a positive axial acceleration of 1.1g. Two nodes 120° apart (one node at the symmetry plane and a second node 120° from the first) were restrained along the bolt diameter at the top of the structural lid in the axial direction. Additionally, the nodes along the centerline of the lids and bottom plate were restrained in the radial direction, and the nodes along the symmetry face were restrained in the direction normal to the symmetry plane.

The resulting maximum stresses in the canister for combined loads are summarized in Tables 3.4.4.1-8, 3.4.4.1-9, and 3.4.4.1-10 for primary membrane, primary membrane plus primary bending, and primary membrane plus primary bending plus secondary stresses, respectively. The sectional stresses at 15 axial locations were obtained for each angular division of the model (a total of 21 angular locations for each axial location). The locations for the stress sections are shown in Figure 3.4.4.1-4. As shown in Tables 3.4.4.1-8 through 3.4.4.1-10, the canister maintains positive margins of safety for the combined load condition.

3.4.4.1.6 Canister Fatigue Evaluation

The purpose of this section is to evaluate the effects of thermal and mechanical cyclic loading conditions on the canister during storage conditions using the criteria presented in ASME Code, Section III, Subsection NB-3222.4 for the canister and Subsection NG-3222.4 for the fuel basket.

During storage conditions, the canister is housed in the vertical concrete storage cask. The storage cask is a shielded reinforced concrete overpack designed to hold a canister during long-term storage conditions. The storage cask is constructed of a thick inner steel liner surrounded by 21 inches of reinforced concrete. Because the carbon steel inner liner will be subjected to a number of temperature/stress loading cycles (1 cycle x 365 days x 50 years = 18,250 cycles) that is less than the minimum number (20,000 cycles) specified for evaluation in Table A-K4.1 of the AISC Manual of Steel Construction, no further fatigue evaluation of the inner liner is required.

Fatigue effects on the canister are addressed using the criteria presented in ASME Section III, Subsection NB-3222.4 and NG-3222.4.

In accordance with these subsections, fatigue analysis need not be performed provided the conditions of six cases are met. The six cases are as follows:

1. Atmospheric to Service Pressure Cycle
2. Normal Service Pressure Fluctuation
3. Temperature Difference — Startup and Shutdown
4. Temperature Difference — Normal Service
5. Temperature Difference — Dissimilar Materials
6. Mechanical Loads

Evaluation of these conditions is presented in the following sections.

Condition 1 — Atmospheric to Service Pressure Cycle

The ASME code requires that the specified number of times that the pressure will be cycled from atmospheric pressure to service pressure and back to atmospheric pressure during normal service does not exceed the number of allowable cycles for the material. In the case of the canister and basket, the cycle from atmospheric to service pressure happens only twice. Since this operation occurs only twice during the 50-year life of the canister (once when the canister is sealed and once when it is opened), atmospheric to service pressure cycle loading of the canister and basket does not cause fatigue failure.

Condition 2 — Normal Service Pressure Fluctuation

To prevent fatigue failure of the canister, the specified full range of pressure fluctuations during normal service must not exceed:

$$P_f = \frac{1}{3} \times P_d \times \left(\frac{S_a}{S_m}\right) = \frac{20 \times 28.2}{3 \times 16.7} = 11.3 \text{ psi},$$

where:

S_a = 28.2 ksi, the value obtained from the design fatigue curve for service cycles < 10^6,

S_m = 16.7 ksi, the allowable stress intensity,

P_d = 20 psi, the design pressure (bounds maximum pressure of 17.93 psi for transfer conditions).

The maximum pressure differential for the canister occurs between transfer and storage conditions. For normal and transfer conditions the maximum pressure differential is:

ΔP = 17.93 - 11.32 = 6.61 psi < 11.3 psi.

Therefore, the effective number of cycles is zero.

Condition 3 — Temperature Difference — Startup and Shutdown

This condition is not applicable. It is only required for power plant startup and shutdown processes.

Condition 4 — Temperature Difference — Normal and Off-Normal Service

Canister Evaluation

The ASME Code specifies that temperature excursions are not significant if the temperature difference between two adjacent points does not change by more than the quantity:

$$\Delta T = \frac{S_a}{2E\alpha} = 58^\circ F,$$

where:

S_a = 28,200 psi, the value obtained from the fatigue curve for service cycles < 10^6,

$E = 27 \times 10^6$ psi, modulus of elasticity at 300 °F,

$\alpha = 9 \times 10^{-6}$ in./in./°F.

For surface temperature differences on surfaces of revolution in the meridional (axial) direction, adjacent points are defined as points that are less than the distance $2\sqrt{Rt}$, where R is the radius measured normal to the surface, from the axis of rotation to the midwall and t is the thickness of the part at the point under consideration. For surface temperature differences on surfaces of revolution in the circumferential direction and on flat parts, such as flanges and flat heads, adjacent points are defined as any two points on the same surface.

The greatest cyclic temperature difference will occur between the off-normal, severe hot (ambient temperature = 100°F) and the off-normal, severe cold (ambient temperature = -40°F) conditions as evaluated in the thermal evaluation. Accident temperature conditions are not applicable.

At the hot condition, the canister bottom plate temperature varies from 237°F at its center to 123°F at its extreme radial point, a ΔT of 114°F. At the cold condition, the canister bottom plate temperature varies from 78.3°F at its center to -24.6°F at its extreme radial point, a ΔT of 102.9°F. Therefore, in cycling from 100°F ambient to -40°F ambient conditions, the ΔT between adjacent points changes by 11.1°F, which is less than the 58°F ΔT and is not considered to be a significant excursion. Heat transfer is uniform around the circumference; therefore, no cyclic ΔT exists in adjacent points on a circumference of the shell.

At the hot condition the canister shell temperature varies from 347.5°F at its center to 121.9°F at its top, a ΔT of 225.6°F. At the cold condition, the canister shell temperature varies from 187.4°F at its center to -25.6°F at its top, a ΔT of 213°F. The distance between adjacent points is:

$$d_p = 2\sqrt{Rt} = 9.35 \text{ in.,}$$

where:

$R = 70.64/2 - 0.625/2 = 35.0$ in, the mean radius of the canister shell,

$t = 0.625$ in., the wall thickness of the canister shell.

At T_{amb}=100 °F, the ΔT between center of canister and end of canister = (347.5°F - 121.9°F) = 225.6°F. The ΔT of adjacent points is (225.6°F / 61.25 in.)(9.35 in.) = 34.4°F.

At T_{amb} = -40 °F, the ΔT between center of canister and end of canister = [187.4°F - (-25.6°F)] = 213°F. The ΔT of adjacent points is (213.0°F / 61.25 in.)(9.35 in.) = 32.5°F.

Therefore, in cycling from 100°F ambient to -40°F ambient conditions, the ΔT between adjacent points changes by 1.9°F, which is less than the 58°F ΔT and is not considered to be a significant excursion.

Basket Evaluation

In storage, the basket is isolated from the influence of environmental temperature excursions by the canister. Any temperature differences within the basket structure are bounded by the evaluation of the temperature differences evaluated for the canister.

Condition 5 — Temperature Difference Between Dissimilar Materials

The canister is constructed of 304L stainless steel and does not contain dissimilar materials. The basket is constructed of several materials. However, all materials except the support disks are free to expand, thus relieving any thermal stress concentration. As noted under the Condition 4 discussion, the temperature differences within the basket are bounded by the temperature differences evaluated for the canister.

Condition 6 — Mechanical Loads

Mechanical loads are not applied to the storage cask and canister during storage conditions. Therefore, no further evaluation is required.

3.4.4.1.7 Canister Pressure Test

The canister is tested using an air over water pressure test in accordance with ASME Code Section III, NB-6221 through NB-6223. The calculated pressure for normal conditions is 7.9 psig. A normal condition design-basis pressure of 11.5 psig is conservatively applied. In accordance with NB-6221, the test pressure applied is 15 psig (≈11.5 x 1.25). The stress resulting from the pressure test is evaluated in accordance with the requirements of NB-3226. The canister is not reused, and the pressure test is conducted only once. Therefore, the pressure test is not considered in the fatigue analysis.

NB-3226 requires that P_m not exceed 0.9 S_y at the test temperature. To show that the canister air over water pressure test meets this condition, the stress intensities calculated for the canister due to the normal condition design pressure of 11.5 psig (Section 3.4.4.1.3), are ratioed to account for the 15 psig test pressure. The canister material is ASME SA-240, Type 304L stainless steel, and the material temperature is conservatively taken to be 250°F. From Table 3.4.4.1-4, the maximum primary stress intensity, P_m, is 4.60 ksi. Therefore:

$$(P_m)_{test} = (15/11.5) \text{ x } (4.60 \text{ ksi}) = 6.0 \text{ ksi, which is} < .9S_y = 18.3 \text{ ksi } (.9 \text{ x } 20.3 \text{ ksi})$$

Thus, the criteria is met.

NB-3226 requires that for $P_m < 0.67S$, the primary membrane plus bending stress intensity, $P_m + P_b$, be $\leq 1.35S_y$. From Table 3.4.4.1-5, $P_m + P_b = 10.02$ ksi. Therefore:

$$(P_m + P_b)_{test} = (15/11.5) \times (10.02 \text{ ksi}) = 13.1 \text{ ksi, which is} \leq 1.35S_y = 27.4 \text{ ksi } (1.35 \times 20.3 \text{ ksi})$$

Therefore, the criteria is met.

The exterior of the canister is at atmospheric pressure at the time the pressure test is conducted, and no external pressure is applied to the canister. Consequently, the evaluation of NB-3133 is not required.

Finally, since the test pressure is equal to the 1.25 times the design pressure, no additional stress calculations are required in accordance with NB-3226, Subparagraph e.

3.4.4.1.8 Fuel Basket Support Disk Evaluation

The response of the fuel basket support disks to storage and handling conditions was evaluated using an ANSYS finite element model that represented a one-quarter section of a single support disk. These loads consist of dead load, handling, and thermal. During storage (dead load) and handling, each support disk supports its own weight and is supported at eight locations by the tie-rod spacers (represented as nodal point restraints in the model). Since all of the support disks experience the same loading conditions during storage and handling, only one support disk was modeled. The support disk model, shown in Figure 3.4.4.1-5 with boundary conditions, was constructed of ANSYS SHELL63 three-dimensional, six degree-of-freedom, elastic shell elements.

The structural analyses of the ANSYS support disk model were performed with temperatures that envelope those experienced by the support disk during storage and handling conditions (100°F and -40°F ambient temperatures). Prior to performing the structural analyses, the steady-state temperature distribution in the support disk model was determined using temperature information from the storage and transfer thermal analyses. This was accomplished by converting the SHELL63 structural elements to SHELL57 thermal elements. The maximum support disk temperature (450°F) was applied to the nodes at the center slot and the minimum support disk temperature (100°F) was applied to the nodes around the outer circumferential edge. All other nodal temperatures were obtained by a steady state conduction solution.

The structural analyses were performed using the SHELL63 structural elements. Since the model represents a one-quarter section of the support disk, in-plane translations and rotations were restrained at the two symmetry faces. Two nodes at the locations of the tie-rod spacers were restrained in the axial direction. The dead load stresses were then calculated by applying a 1.1g acceleration to the entire model in the axial direction, and the handling stresses were calculated by applying a 1.1g acceleration to the entire model in the axial direction. Thermal stresses were also evaluated in addition to both dead load and handling load. The results of the support disk structural analyses for dead load, handling load, and thermal load are presented in Table 3.4.4.1-11. As shown in Table 3.4.4.1-11, the support disk maintains positive margins of safety for the conditions analyzed.

3.4.4.1.9 Fuel Basket Weldments Evaluation

The response of the fuel basket top and bottom weldments to storage and handling conditions was analyzed using ANSYS finite element models representing one-quarter section of a top and a bottom weldment. These loads consist of the dead weight and handling loads and thermal expansion. During storage (dead load) and handling, the top weldment plate supports its own weight, the weight of eight structural ribs, and the weight of a circumferential ring, which is welded to the plate. The top weldment plate is supported at eight locations by the tie-rod spacers (represented as nodal point restraints in the model). During storage (dead load) and handling, the bottom weldment plate supports its own weight plus the weight of 36 fuel tubes applied as sets of nodal forces around the slot locations. The bottom weldment plate is supported at eight locations by the tie-rod spacers and at 12 locations by structural ribs (represented as nodal point restraints in the model). The top and bottom weldments are both constructed of SA240, Type 304 stainless steel. The top and bottom weldments model, shown in Figures 3.4.4.1-6 and 3.4.4.1-7, respectively, with boundary conditions, were constructed of ANSYS SHELL63 three-dimensional, six degree-of-freedom, elastic shell elements.

The structural analyses of the ANSYS weldment models were performed using the methodology described in Section 3.4.4.1.8. Temperatures employed for the thermal conduction analysis of the weldments are shown below:

Weldment	Maximum Temperature (°F) (at center)	Minimum Temperature (°F) (at circumference)
Top	400	380
Bottom	150	100

Since the ANSYS finite element models represent a one-quarter section of each weldment, in-plane translations and rotations were restrained at the plane of symmetry face. In each weldment model, two nodes at the locations of the tie-rod spacers were restrained in the axial direction. In addition, for the bottom weldment model, two nodes at the location of the support pads were restrained in the axial direction. The dead load stresses were then calculated by applying a 1g acceleration to the entire model in the axial direction, and the handling stresses were calculated by applying a 1.1g acceleration to the entire model in the axial direction. Thermal stresses were analyses for dead load, handling load, and thermal load are presented in Table 3.4.4.1-11. To account for the hottest temperatures experienced by the weldments during storage and handling, the allowable stresses are taken at 658°F for the top weldment and 662°F for the bottom weldment. As shown in Table 3.4.4.1-11, the weldments maintain positive margins of safety for the combined load conditions.

3.4.4.1.10 Fuel Tube Analysis

The fuel tube provides a sealed cavity to mount BORAL poison plates within the fuel basket structure but the fuel tube does not provide structural support of the fuel assembly. The fuel tube design is presented in Figure 3.4.4.1-8. The thickness of the tube wall is 0.048 inch. A structural evaluation of the tube has been performed for the dead load and handling load conditions. The thermal stress is considered to be negligible since the tube is free to expand in both axial and radial directions. The handling load is considered to be 10% of the dead load.

During storage, the fuel assemblies are in contact with the bottom weldment, which is supported by the canister bottom plate. In the vertical position, the fuel assembly load is not carried by the fuel tubes. The fuel tubes are supported by the bottom weldment. Therefore, evaluation of the fuel tube is performed considering the weight of the fuel tube, with a g-load of 1.1 (to account for both the dead load and handling load) carried by the tube cross-section. From the dimensions of the tube shown in Figure 3.4.4.1-8, the cross sectional area is:

$$\begin{aligned}\text{Area} &= (7.8 + 2 \times 0.048)^2 - 7.8^2 \\ &= 1.507 \text{ in}^2\end{aligned}$$

The weight of a fuel tube, including the BORAL plates, is 78 pounds. Considering a g-load of 1.1, the maximum compressive and bearing stress in the fuel tube is 57 psi (78 x 1.1 / 1.507). Limiting the compressive stress level in the tube to the material yield strength ensures the tube remains in position in storage conditions. The yield strength of Type 304 stainless steel is 17,300 psi at a conservatively high temperature of 750°F.

$$\begin{aligned}\text{Margin of Safety} &= 17{,}300/57 - 1 \\ &= +\text{Large}\end{aligned}$$

Figure 3.4.4.1-1 Canister ANSYS Finite Element Model

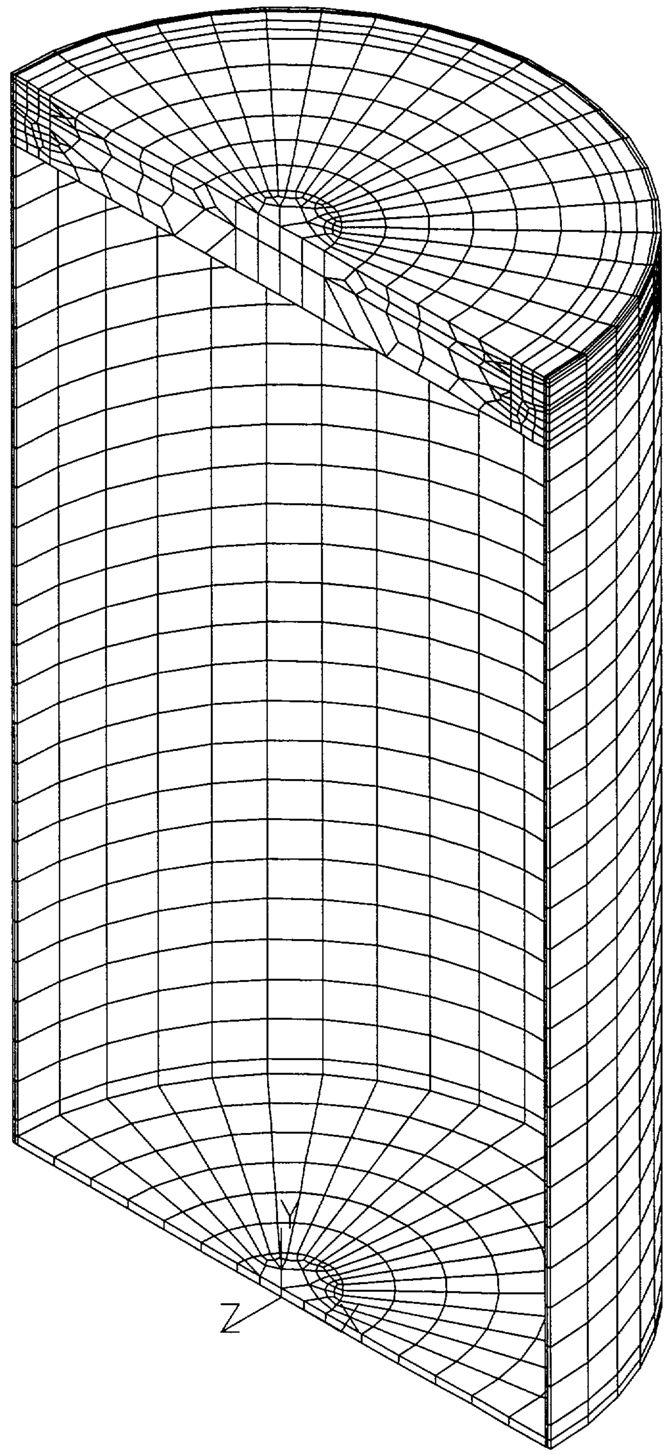


Figure 3.4.4.1-2 Weld Regions of Canister ANSYS Finite Element Model at Structural and Shield Lids

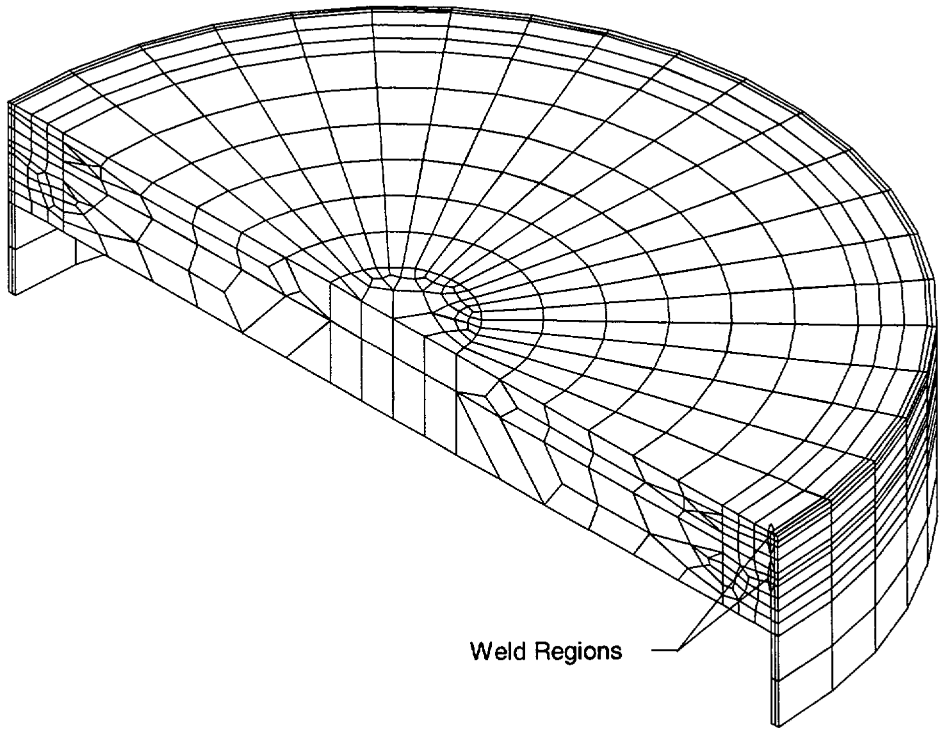


Figure 3.4.4.1-3 Bottom Plate of the Canister ANSYS Finite Element Model

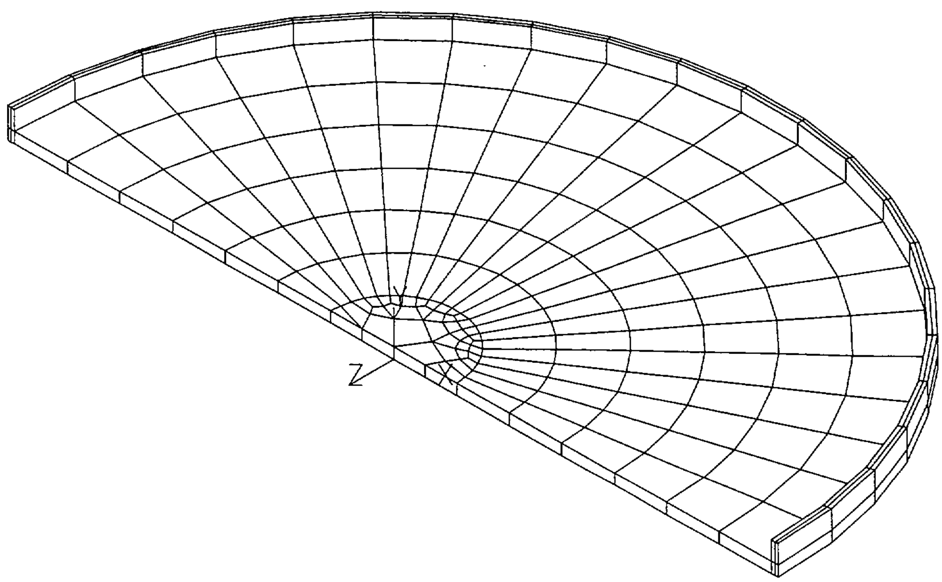
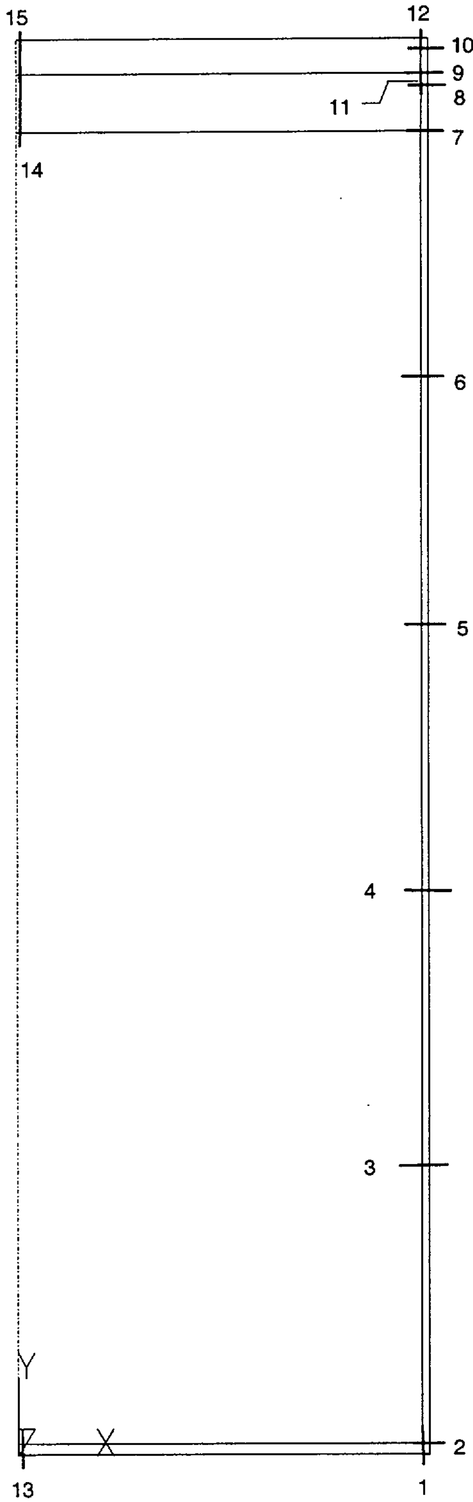
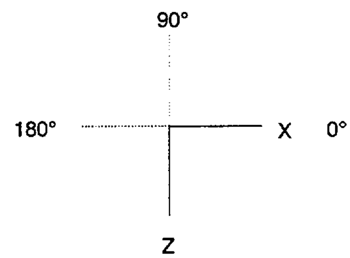


Figure 3.4.4.1-4 Locations for Section Stresses in the Canister ANSYS Finite Element Model





Top View of Axis

Location of sections at Z=0 (Note: these sections are repeated at each angular division in the 180° model). All units are in inches.

	Node 1		Node 2	
Loc.	X1	Y1	X2	Y2
1	34.695	0.0	34.695	1.0
2	34.695	1.0	35.32	1.0
3	34.695	23.7	35.32	23.7
4	34.695	46.4	35.32	46.4
5	34.695	69.1	35.32	69.1
6	34.695	91.8	35.32	91.8
7	34.695	114.5	35.32	114.5
8	34.695	118.5	35.32	118.5
9	34.695	119.5	35.32	119.5
10	34.695	121.62	35.32	121.62
11	34.695	118.5	34.695	119.5
12	34.695	119.5	34.695	122.5
13	0.01	0.0	0.01	1.0
14	0.01	114.5	0.01	119.5
15	0.01	119.5	0.01	122.5

Figure 3.4.4.1-5 Fuel Basket Support Disk ANSYS Finite Element Model

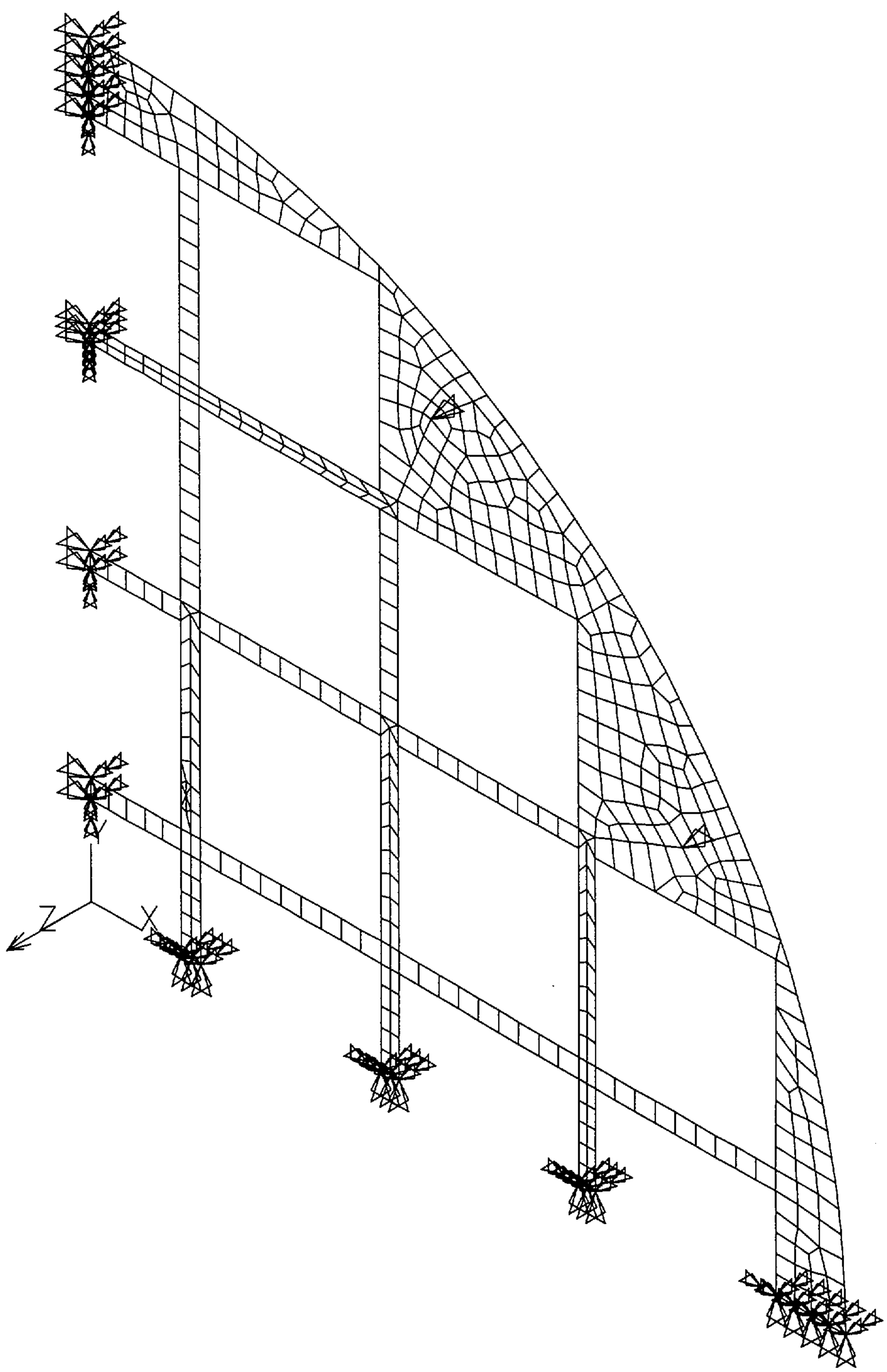


Figure 3.4.4.1-6 Fuel Basket Top Weldment ANSYS Finite Element Model

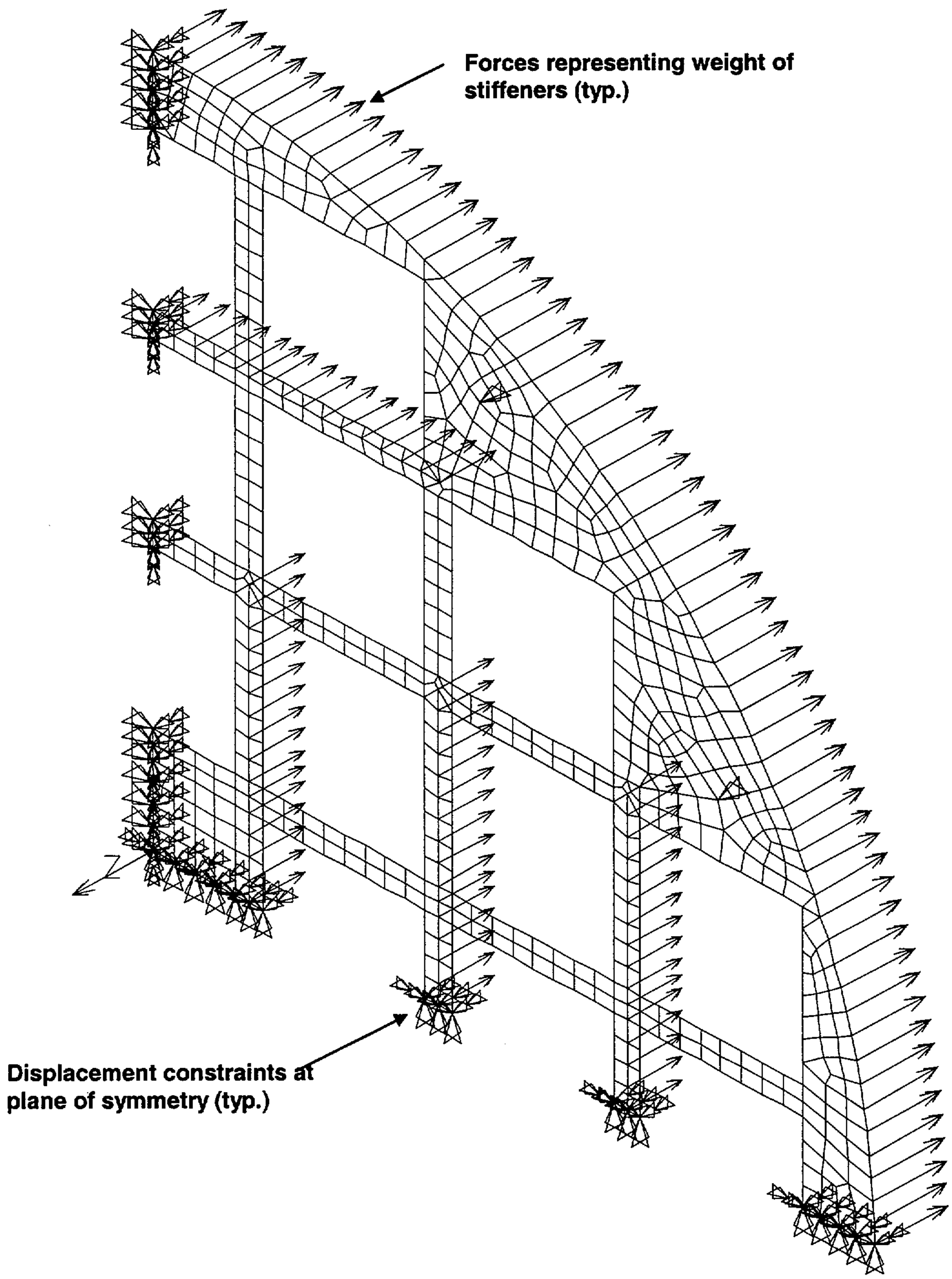


Figure 3.4.4.1-7 Fuel Basket Bottom Weldment ANSYS Finite Element Model

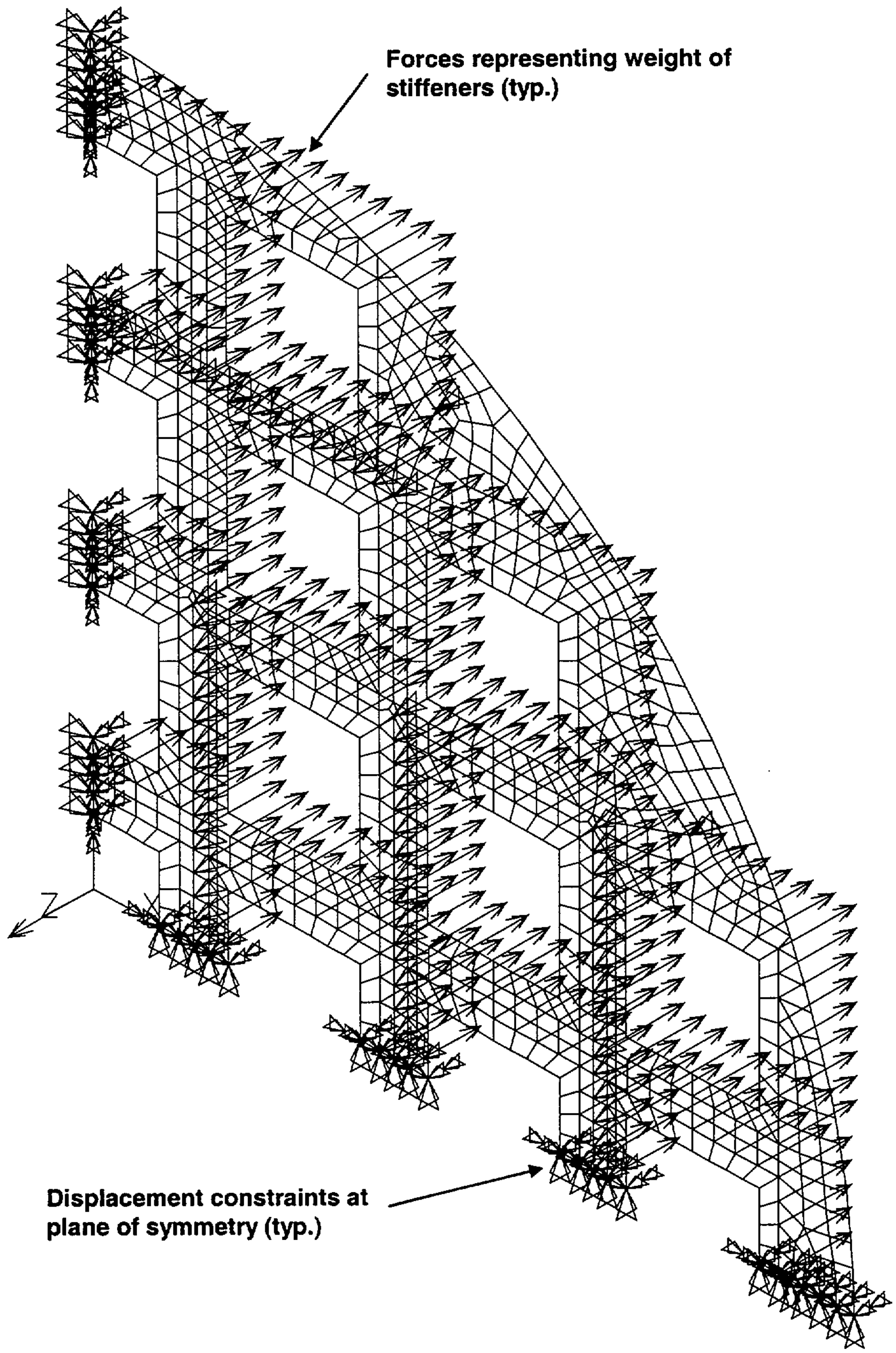


Figure 3.4.4.1-8 Fuel Tube Configuration

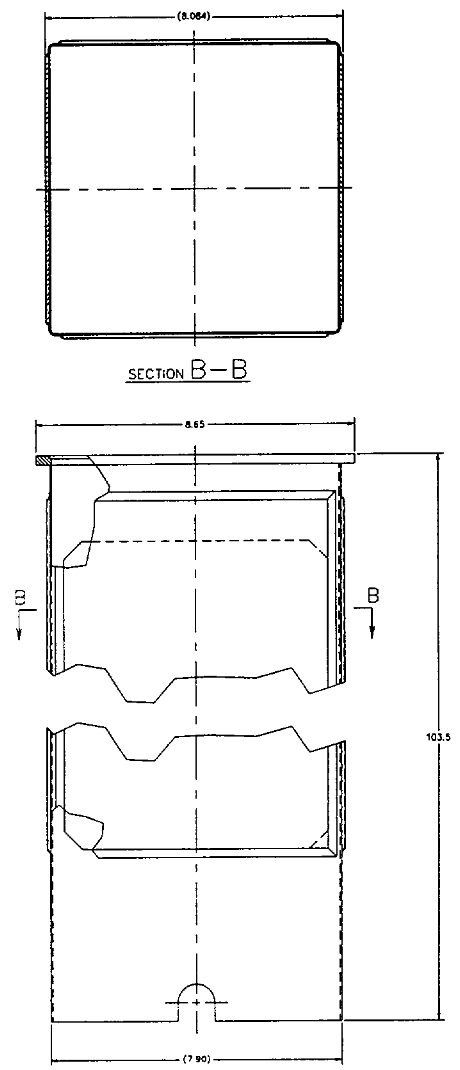


Table 3.4.4.1-1 Summary of Maximum Canister Thermal Stresses (ksi)

Location No.[1]	SX	SY	SZ	SXY	SYZ	SXZ	Stress Intensity
1	0.2	0.9	3.2	0.1	< 0.1	-0.2	3.08
2	-0.3	-1.8	2.0	-0.1	< 0.1	0.2	3.73
3	< 0.1	< 0.1	< 0.1	< 0.1	< 0.1	< 0.1	0.04
4	-0.1	0.1	0.3	< 0.1	< 0.1	< 0.1	0.34
5	< 0.1	-0.1	0.1	< 0.1	< 0.1	< 0.1	0.26
6	< 0.1	< 0.1	< 0.1	< 0.1	< 0.1	< 0.1	0.04
7	< 0.1	0.5	-0.8	-0.1	< 0.1	-0.1	1.37
8	1.2	-2.7	-0.7	1.1	-0.1	-0.1	4.43
9	-2.0	9.7	2.2	0.4	< 0.1	0.3	11.78
10	2.5	-10.7	-1.6	0.7	-0.1	-0.3	13.29
11	-4.7	-5.1	-2.7	-0.7	< 0.1	0.1	2.90
12	-4.3	1.6	< 0.1	-0.8	-0.1	-0.3	6.12
13	-19.5	-6.2	-18.8	< 0.1	-0.6	< 0.1	13.35
14	2.4	4.0	2.5	< 0.1	-0.2	< 0.1	1.64
15	-7.1	-5.0	-6.9	< 0.1	0.3	< 0.1	2.17

[1] See Figure 3.4.4.1-4 for definition of locations of stress sections.

Table 3.4.4.1-2 Summary of Maximum Canister Dead Load Primary Membrane (P_m) Stresses (ksi)

Location No.[1]	SX	SY	SZ	SXY	SYZ	SXZ	Stress Intensity
1	< 0.1	< 0.1	< 0.1	< 0.1	< 0.1	< 0.1	0.04
2	< 0.1	-0.1	< 0.1	< 0.1	< 0.1	< 0.1	0.09
3	< 0.1	-0.1	< 0.1	< 0.1	< 0.1	< 0.1	0.09
4	< 0.1	-0.1	< 0.1	< 0.1	< 0.1	< 0.1	0.09
5	< 0.1	-0.1	< 0.1	< 0.1	< 0.1	< 0.1	0.08
6	< 0.1	-0.1	< 0.1	< 0.1	< 0.1	< 0.1	0.07
7	< 0.1	-0.1	< 0.1	< 0.1	< 0.1	< 0.1	0.05
8	< 0.1	< 0.1	< 0.1	< 0.1	< 0.1	< 0.1	0.03
9	< 0.1	< 0.1	< 0.1	< 0.1	< 0.1	< 0.1	0.08
10	< 0.1	< 0.1	< 0.1	< 0.1	< 0.1	< 0.1	0.07
11	< 0.1	< 0.1	< 0.1	< 0.1	< 0.1	< 0.1	0.04
12	< 0.1	-0.1	< 0.1	< 0.1	< 0.1	< 0.1	0.06
13	< 0.1	< 0.1	< 0.1	< 0.1	< 0.1	< 0.1	0.01
14	< 0.1	< 0.1	< 0.1	< 0.1	< 0.1	< 0.1	0.02
15	< 0.1	< 0.1	< 0.1	< 0.1	< 0.1	< 0.1	0.01

[1] See Figure 3.4.4.1-4 for definition of locations of stress sections.

Table 3.4.4.1-3 Summary of Maximum Canister Dead Load Primary Membrane Plus Primary Bending ($P_m + P_b$) Stresses (ksi)

Location No.[1]	SX	SY	SZ	SXY	SYZ	SXZ	Stress Intensity
1	< 0.1	< 0.1	< 0.1	< 0.1	< 0.1	< 0.1	0.06
2	< 0.1	-0.1	< 0.1	< 0.1	< 0.1	< 0.1	0.12
3	< 0.1	-0.1	< 0.1	< 0.1	< 0.1	< 0.1	0.09
4	< 0.1	-0.1	< 0.1	< 0.1	< 0.1	< 0.1	0.09
5	< 0.1	-0.1	< 0.1	< 0.1	< 0.1	< 0.1	0.08
6	< 0.1	-0.1	< 0.1	< 0.1	< 0.1	< 0.1	0.07
7	< 0.1	-0.1	< 0.1	< 0.1	< 0.1	< 0.1	0.07
8	< 0.1	< 0.1	< 0.1	< 0.1	< 0.1	< 0.1	0.05
9	< 0.1	-0.2	-0.1	< 0.1	< 0.1	< 0.1	0.20
10	< 0.1	0.1	< 0.1	< 0.1	< 0.1	< 0.1	0.18
11	0.1	0.1	< 0.1	< 0.1	< 0.1	< 0.1	0.08
12	0.1	< 0.1	< 0.1	< 0.1	< 0.1	< 0.1	0.09
13	< 0.1	< 0.1	< 0.1	< 0.1	< 0.1	< 0.1	0.01
14	0.1	< 0.1	0.1	< 0.1	< 0.1	< 0.1	0.09
15	-0.1	< 0.1	-0.1	< 0.1	< 0.1	< 0.1	0.06

[1] See Figure 3.4.4.1-4 for definition of locations of stress sections.

Table 3.4.4.1-4 Summary of Maximum Canister Internal Pressure Load Primary Membrane (P_m) Stresses (ksi)

Location No.[1]	SX	SY	SZ	SXY	SYZ	SXZ	Stress Intensity
1	-0.6	3.0	0.9	0.7	0.3	-0.1	3.92
2	1.6	-1.2	-1.0	0.6	0.1	0.1	3.02
3	< 0.1	0.3	0.6	< 0.1	< 0.1	< 0.1	0.62
4	< 0.1	0.3	0.6	< 0.1	< 0.1	< 0.1	0.64
5	< 0.1	0.3	0.6	< 0.1	< 0.1	< 0.1	0.64
6	< 0.1	0.3	0.6	< 0.1	< 0.1	< 0.1	0.64
7	< 0.1	0.3	0.3	< 0.1	< 0.1	< 0.1	0.33
8	0.1	0.2	0.2	0.1	< 0.1	< 0.1	0.18
9	-0.2	0.2	0.1	< 0.1	< 0.1	< 0.1	0.41
10	0.2	-0.1	0.1	< 0.1	< 0.1	< 0.1	0.35
11	< 0.1	-0.1	0.1	< 0.1	< 0.1	< 0.1	0.20
12	< 0.1	0.3	0.2	< 0.1	< 0.1	< 0.1	0.31
13	0.7	< 0.1	0.7	-0.7	-2.2	< 0.1	4.60
14	< 0.1	< 0.1	< 0.1	< 0.1	< 0.1	< 0.1	0.06
15	< 0.1	< 0.1	< 0.1	< 0.1	< 0.1	< 0.1	0.07

[1] See Figure 3.4.4.1-4 for definition of locations of stress sections.

Table 3.4.4.1-5 Summary of Maximum Canister Internal Pressure Load Primary Membrane Plus Primary Bending ($P_m + P_b$) Stresses (ksi)

Location No.[1]	SX	SY	SZ	SXY	SYZ	SXZ	Stress Intensity
1	-4.2	0.3	0.6	1.0	0.4	-0.2	5.29
2	0.7	-9.2	-3.4	0.7	< 0.1	0.3	10.02
3	< 0.1	0.4	0.6	< 0.1	< 0.1	-0.1	0.66
4	< 0.1	0.3	0.6	< 0.1	< 0.1	< 0.1	0.65
5	< 0.1	0.3	0.6	< 0.1	< 0.1	< 0.1	0.65
6	< 0.1	0.3	0.6	< 0.1	< 0.1	< 0.1	0.65
7	< 0.1	0.4	0.3	< 0.1	< 0.1	< 0.1	0.39
8	0.1	0.1	0.2	-0.1	< 0.1	< 0.1	0.21
9	-0.1	0.9	0.4	< 0.1	< 0.1	< 0.1	1.03
10	0.2	-0.7	-0.1	< 0.1	< 0.1	< 0.1	0.92
11	-0.3	-0.4	< 0.1	0.1	< 0.1	< 0.1	0.42
12	-0.3	0.1	0.1	-0.1	< 0.1	< 0.1	0.47
13	10.3	1.8	10.0	-0.7	-2.2	< 0.1	9.40
14	-0.6	-0.1	-0.5	< 0.1	< 0.1	< 0.1	0.44
15	0.3	< 0.1	0.3	< 0.1	< 0.1	< 0.1	0.31

[1] See Figure 3.4.4.1-4 for definition of locations of stress sections.

Table 3.4.4.1-6 Summary of Maximum Canister Dead Load + Handling Load Primary Membrane (P_m) Stresses (ksi)

Location No.[1]	SX	SY	SZ	SXY	SYZ	SXZ	Stress Intensity
1	-0.7	3.1	0.9	-0.7	0.3	0.1	4.11
2	1.6	-1.3	-1.1	-0.6	0.1	-0.2	3.15
3	< 0.1	0.4	< 0.1	< 0.1	< 0.1	< 0.1	0.41
4	< 0.1	0.4	< 0.1	< 0.1	< 0.1	< 0.1	0.45
5	< 0.1	0.5	< 0.1	< 0.1	< 0.1	< 0.1	0.52
6	< 0.1	0.6	< 0.1	< 0.1	< 0.1	< 0.1	0.64
7	< 0.1	0.9	< 0.1	< 0.1	0.1	< 0.1	0.94
8	< 0.1	0.9	0.2	< 0.1	0.1	< 0.1	0.90
9	-0.2	1.2	0.3	< 0.1	0.1	< 0.1	1.34
10	-0.3	0.6	0.6	-0.3	0.1	0.1	1.08
11	-0.1	0.8	0.2	< 0.1	0.1	< 0.1	0.90
12	0.2	< 0.1	0.8	-0.3	< 0.1	0.1	1.07
13	0.7	< 0.1	0.7	-0.7	-2.3	< 0.1	4.86
14	< 0.1	< 0.1	< 0.1	< 0.1	< 0.1	< 0.1	0.03
15	< 0.1	< 0.1	< 0.1	< 0.1	< 0.1	< 0.1	0.05

[1] See Figure 3.4.4.1-4 for definition of locations of stress sections.

Table 3.4.4.1-7 Summary of Maximum Canister Dead Load + Handling Load Primary Membrane Plus Primary Bending ($P_m + P_b$) Stresses (ksi)

Location No.[1]	SX	SY	SZ	SXY	SYZ	SXZ	Stress Intensity
1	-4.3	0.3	0.7	-1.0	0.4	0.2	5.50
2	0.7	-9.7	-3.6	-0.8	< 0.1	-0.3	10.51
3	< 0.1	0.5	-0.1	< 0.1	< 0.1	< 0.1	0.53
4	< 0.1	0.4	-0.1	< 0.1	< 0.1	< 0.1	0.52
5	< 0.1	0.5	-0.1	< 0.1	< 0.1	< 0.1	0.62
6	< 0.1	0.6	-0.2	< 0.1	< 0.1	< 0.1	0.74
7	< 0.1	1.0	< 0.1	< 0.1	< 0.1	< 0.1	1.01
8	< 0.1	1.1	0.3	0.1	0.1	< 0.1	1.08
9	0.2	1.5	-0.1	< 0.1	-0.1	-0.3	1.73
10	-0.4	0.9	0.6	-0.5	0.1	0.1	1.70
11	0.2	1.2	-0.1	< 0.1	< 0.1	-0.2	1.37
12	0.7	-0.4	1.0	-0.1	-0.1	0.2	1.56
13	10.9	1.9	10.5	-0.7	-2.3	< 0.1	9.91
14	-0.2	< 0.1	-0.2	< 0.1	< 0.1	< 0.1	0.18
15	0.2	< 0.1	0.2	< 0.1	< 0.1	< 0.1	0.18

[1] See Figure 3.4.4.1-4 for definition of locations of stress sections.

Table 3.4.4.1-8 Summary of Maximum Canister Combined Load Primary Membrane (P_m) Stresses (ksi)

Location No.[1]	SX	SY	SZ	SXY	SYZ	SXZ	Stress Intensity	Allowable Stress[2]	Margin of Safety
1	-1.3	6.1	1.9	-1.4	0.6	0.2	8.02	16.70	1.08
2	3.2	-2.5	-2.1	-1.2	0.2	-0.3	6.16	16.70	1.71
3	< 0.1	0.7	0.6	< 0.1	< 0.1	< 0.1	0.71	14.50	19.42
4	< 0.1	0.8	0.6	< 0.1	< 0.1	< 0.1	0.76	14.50	18.08
5	< 0.1	0.8	0.6	< 0.1	< 0.1	< 0.1	0.84	14.50	16.26
6	< 0.1	0.9	0.6	< 0.1	< 0.1	0.1	0.94	14.50	14.43
7	< 0.1	1.2	0.3	< 0.1	0.1	< 0.1	1.23	16.70	12.63
8	0.1	1.1	0.4	0.1	0.1	< 0.1	1.06	16.70	14.77
9	-0.3	1.3	0.4	< 0.1	0.1	0.1	1.67	16.70	9.00
10	< 0.1	0.5	0.8	-0.2	< 0.1	0.1	0.86	16.70	18.31
11	-0.1	0.9	0.3	< 0.1	0.1	0.1	0.98	16.70	15.97
12	0.1	0.3	1.1	-0.3	< 0.1	0.1	1.19	10.69[4]	7.98
13	1.4	0.1	1.3	-1.5	-4.5	< 0.1	9.47	16.70	0.76
14[3]	-0.1	< 0.1	-0.1	< 0.1	< 0.1	< 0.1	0.09	20.00	221.22
15	< 0.1	< 0.1	< 0.1	< 0.1	0.1	< 0.1	0.14	16.70	121.34

1 See Figure 3.4.4.1-4 for definition of locations of stress sections.

2 Allowable stresses for bottom plate region (location no. 1-2, 13) taken at 250°F; allowable stresses for canister shell region between shield lid and bottom plate (location no. 3-6) taken at 550°F; allowable stresses for structural/shield lid region (location no. 7-12, 14-15) taken at 250°F.

3 The allowable stress for SA240, Type 304 stainless steel was used for location No. 14. The allowable stress for SA240, Type 304L stainless steel was used for all other locations.

4 Includes two stress reduction factors for weld = 0.8 x 0.8 = 0.64 (See Section 3.6.1).

Table 3.4.4.1-9 Summary of Maximum Canister Combined Load Primary Membrane Plus Primary Bending ($P_m + P_b$) Stresses (ksi)

Location No.[1]	SX	SY	SZ	SXY	SYZ	SXZ	Stress Intensity	Allowable Stress[2]	Margin of Safety
1	-8.5	0.7	1.3	-2.0	0.8	0.4	10.77	25.05	1.33
2	1.4	-18.8	-7.0	-1.5	< 0.1	-0.6	20.50	25.05	0.22
3	< 0.1	0.9	0.6	< 0.1	< 0.1	< 0.1	0.92	21.75	22.64
4	< 0.1	0.8	0.7	< 0.1	< 0.1	0.1	0.79	21.75	26.53
5	< 0.1	0.9	0.8	< 0.1	< 0.1	0.1	0.88	21.75	23.72
6	< 0.1	1.0	0.8	< 0.1	< 0.1	0.1	1.00	21.75	20.75
7	< 0.1	1.3	0.3	< 0.1	< 0.1	< 0.1	1.31	25.05	18.14
8	0.1	1.2	0.4	0.1	0.1	< 0.1	1.17	25.05	20.45
9	-0.4	1.4	0.4	< 0.1	0.1	0.1	1.85	25.05	12.53
10	-0.1	1.7	1.1	-0.5	0.1	0.1	2.06	25.05	11.14
11	-0.4	1.0	0.3	< 0.1	0.1	0.1	1.45	25.05	16.26
12	0.3	-0.3	1.3	-0.1	-0.1	0.2	1.66	16.03[4]	8.66
13	21.2	3.7	20.5	-1.4	-4.5	< 0.1	19.33	25.05	0.30
14[3]	-0.5	-0.1	-0.5	< 0.1	< 0.1	< 0.1	0.43	30.00	68.77
15	0.8	0.1	0.8	< 0.1	0.1	< 0.1	0.67	25.05	36.12

1 See Figure 3.4.4.1-4 for definition of locations of stress sections.

2 Allowable stresses for bottom plate region (location no. 1-2, 13) taken at 250°F; allowable stresses for canister shell region between shield lid and bottom plate (location no. 3-6) taken at 550°F; allowable stresses for structural/shield lid region (location no. 7-12, 14-15) taken at 250°F.

3 The allowable stress for SA240, Type 304 stainless steel was used for location no. 14. The allowable stress for SA240, Type 304L stainless steel was used for all other locations.

4 Includes two stress reduction factors for weld = 0.8 x 0.8 = 0.64 (See Section 3.6.1).

Table 3.4.4.1-10 Summary of Maximum Canister Combined Load Primary Membrane Plus Primary Bending Plus Secondary ($P_m + P_b + Q$) Stresses (ksi)

Location No.[1]	SX	SY	SZ	SXY	SYZ	SXZ	Stress Intensity	Allowable Stress[2]	Margin of Safety
1	-9.1	0.6	3.6	2.1	0.8	-0.6	13.38	50.10	2.74
2	2.1	-22.9	-5.4	-1.5	0.1	-0.5	25.20	50.10	0.99
3	< 0.1	0.9	0.6	< 0.1	< 0.1	< 0.1	0.90	43.50	47.33
4	-0.1	0.7	1.0	< 0.1	< 0.1	-0.1	1.11	43.50	38.19
5	-0.1	0.7	0.9	< 0.1	< 0.1	-0.1	1.01	43.50	42.07
6	< 0.1	1.0	0.8	< 0.1	< 0.1	0.1	1.02	43.50	41.65
7	< 0.1	1.8	-0.6	-0.1	0.1	-0.1	2.43	50.10	19.59
8	1.2	-2.3	-0.5	-1.2	-0.1	0.1	4.22	50.10	10.86
9	-2.3	10.8	2.6	0.3	0.1	0.3	13.14	50.10	2.81
10	2.8	-12.4	-1.8	-0.8	-0.2	0.7	15.32	50.10	2.27
11	-5.0	-5.5	-2.8	0.9	< 0.1	-0.1	3.34	50.10	14.01
12	-4.9	1.8	< 0.1	-0.9	-0.1	-0.3	6.99	32.06[4]	3.59
13	-26.4	0.5	-25.2	-1.4	-4.0	< 0.1	27.74	50.10	0.81
14[3]	1.7	3.9	1.8	< 0.1	-0.1	0.1	2.21	60.00	26.15
15	1.7	3.9	1.8	< 0.1	-0.1	0.1	2.21	50.10	21.68

1 See Figure 3.4.4.1-4 for definition of locations of stress sections.

2 Allowable stresses for bottom plate region (location no. 1-2, 13) taken at 250°F; allowable stresses for canister shell region between shield lid and bottom plate (location no. 3-6) taken at 550°F; allowable stresses for structural/shield lid region (location no. 7-12, 14-15) taken at 250°F.

3 The allowable stress for SA240, Type 304 stainless steel was used for location no. 14. The allowable stress for SA240, Type 304L stainless steel was used for all other locations.

4 Includes two stress reduction factors for weld = 0.8 x 0.8 = 0.64 (See Section 3.6.1).

Table 3.4.4.1-11 Summary of Maximum Stresses for the Fuel Basket Weldments and Support Disks

Component	Load Condition	Stress Intensity		Allowable Stress		Margin of Safety
		Reported	Value (psi)	Criteria	Value (psi)	
Top Weldment	Dead Load	P_m	0	S_m	16,299	---
		$P_m + P_b$	3,297	$1.5S_m$	24,449	6.42
	Dead Load + Thermal	$P_m + P_b + Q$	32,364	$3.0S_m$	48,898	0.51
	Dead Load + Handling	P_m	0	S_m	16,299	---
		$P_m + P_b$	3,626	$1.5S_m$	24,449	5.74
	Dead Load + Handling + Thermal	$P_m + P_b + Q$	32,521	$3.0S_m$	48,898	0.50
Bottom Weldment	Dead Load	P_m	0	S_m	16,269	---
		$P_m + P_b$	857	$1.5S_m$	24,403	27.47
	Dead Load + Thermal	$P_m + P_b + Q$	44,094	$3.0S_m$	48,806	0.11
	Dead Load + Handling	P_m	0	S_m	16,269	---
		$P_m + P_b$	942	$1.5S_m$	24,403	24.89
	Dead Load + Handling + Thermal	$P_m + P_b + Q$	44,119	$3.0S_m$	48,806	0.11
Support Disks	Dead Load	P_m	0	S_m	41,528	---
		$P_m + P_b$	870	$1.5S_m$	62,292	70.60
	Dead Load + Thermal	$P_m + P_b + Q$	35,427	$3.0S_m$	124,584	2.52
	Dead Load + Handling	P_m	0	S_m	41,528	---
		$P_m + P_b$	958	$1.5S_m$	62,292	64.02
	Dead Load + Handling + Thermal	$P_m + P_b + Q$	35,495	$3.0S_m$	124,584	2.51

3.4.4.2 Vertical Concrete Storage Cask - Concrete Stress Analysis

This section evaluates the stresses in the storage cask concrete for normal conditions of storage. The evaluation for the steel pedestal at the bottom of the cask is presented in Section 3.4.3.1. The stresses in the concrete due to dead load, live load, and thermal load are calculated below. The evaluations for off-normal and accident loading conditions are presented in Chapter 11. Summary of calculated stresses for the load combinations defined in Table 2.2-2 is presented in Table 3.4.4.2-1. The maximum stress in the concrete and the maximum force in the reinforcing bars and the comparison to their allowable limits are summarized in Table 3.4.4.2-2. As shown in Table 3.4.4.2-2, the storage cask meets the structural requirements of ACI-349-85.

3.4.4.2.1 Dead Load

The dead load of the storage cask concrete is reacted by the lower concrete surface only. The concrete compression stress due to the self-weight of the storage cask is:

$$\sigma_v = -W/A = -21.44 \text{ psi (compression)}$$

where:

W	= 151,364 lbs concrete cask dead weight
D	= 128 in. concrete exterior diameter
ID	= 86 in. concrete interior diameter
A	$= \pi (D^2 - ID^2)/4 =$ 7,059.2 in.^{2}

Stress evaluation at the base of the concrete conservatively considers the weight of the empty concrete cask, rather than the concrete alone. The weight of the canister is not supported by the concrete.

3.4.4.2.2 Live Load

The storage cask is subjected to two live loads: (1) the snow load and (2) the weight of the fully loaded transfer cask resting atop the storage cask. These loads are conservatively assumed to be applied to the concrete portion of the storage cask. No loads are assumed to be taken by the steel liner. The loads from the canister and its contents are transferred to the steel support inside the storage cask and are not applied to the concrete. The stress in the steel support is evaluated in Section 3.4.3.1. Under these conditions, the only stress component is the vertical compression stress.

Snow Load

The snow load on the storage cask is determined in accordance with ANSI/ASCE 7-93 as follows:

The uniformly distributed snow load on the top of the storage cask, P_f, is:

$$P_f = 0.70\, C_e\, C_t\, I\, P_g = 100.8 \text{ lbf/ft}^2 \text{ (Section 2.2.4)}$$

The storage cask top area:

$$A_{top} = \pi\,(D/2)^2 = 12{,}868 \text{ in.}^2 = 89.36 \text{ ft}^2$$

The maximum snow load, F_s, is:

$$F_s = P_f \times A_{top} = 100.8\,(89.36) = 9{,}007 \text{ lbf.}$$

The snow load is uniformly distributed over the top surface of the concrete.

The live load of the transfer cask is 135,473 lbs, which is much greater than the weight of the snow. Consequently, the stress due to the snow load is bounded by the weight of the transfer cask.

W = 135,473 lbs-transfer cask weight (fully loaded)
D = 128 in.-concrete exterior diameter
ID = 86 in.-concrete interior diameter

$$A = \pi\,(D^2 - ID^2)/4 = 7{,}059.2 \text{ in.}^2$$

Compression stress at the base of the concrete is:

$$\sigma_v = W/A = -19.2 \text{ psi (compressive)}$$

3.4.4.2.3 Thermal Load

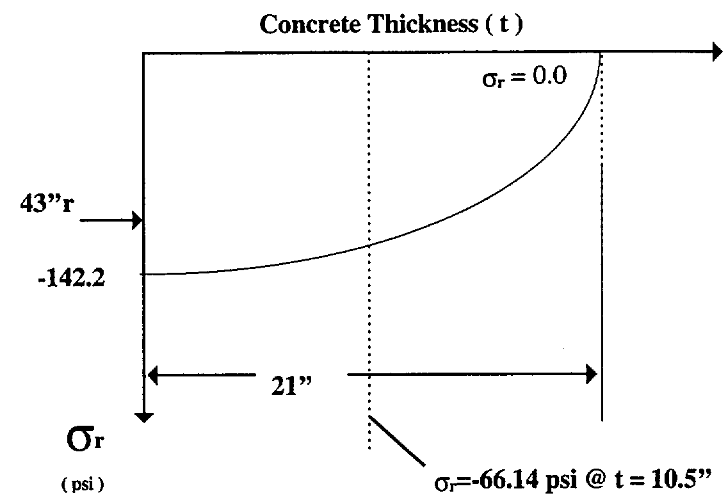
An axisymmetric finite element model consisting of ANSYS PLANE42 elements for the steel liner and concrete shell was developed to calculate the thermal stresses in the concrete (see Fig. 3.4.4.2-1). The nodes at the steel liner/concrete interface are coincident and are connected analytically by coupling the degrees of freedom. Overall shell height is 160 in., the inner radius of the 3.5-in. thick carbon steel liner is 39.5 in., and the outer concrete radius is 64.0 in. The model obtains first a thermal solution (temperatures) and then a structural solution (stresses).

The steady-state, two-dimensional heat transfer conduction solution uses the surface temperature boundary conditions as calculated by the thermal analysis for normal conditions as presented in Section 4.4.1.1. These temperatures were applied without load factor along the steel liner interior and concrete exterior. The coincident nodes located along the steel and concrete interface were coupled with the temperature degree of freedom.

After the thermal solution, the thermal model is converted to a structural model. The nodal temperatures developed from the heat transfer analysis become the thermal load boundary conditions for the structural model.

Analysis with these boundary conditions provides the magnitude of three stress states σ_r, σ_y, σ_θ, which are denoted radial, vertical, and circumferential stresses, respectively (these designations correspond to the x, y, and z axes, respectively, in the model). Stress magnitudes are calculated at various points along the concrete wall mid-span to determine the critical bounding cross sections.

The radial stress (σ_r) varies through the concrete wall as shown in the following diagram.



The maximum interior stress of -142.2 psi is a bearing (compressive) stress due to thermal expansion of the steel liner.

Applying the ACI 349-85 load reduction factor, the allowable bearing stress on the concrete is:

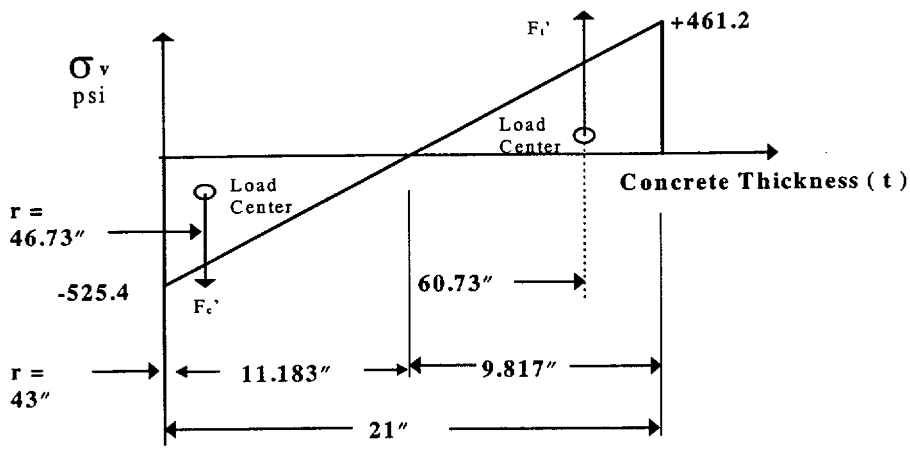
$\phi = 0.70$

$f_c' = 4{,}000$ psi

$\sigma_{bearing} = \phi f_c' = (0.70)(4{,}000) = 2{,}800$ psi

The maximum 75°F normal operating thermally induced stress of -142.2 psi, when factored by the 1.275 load factor (see Table 2.2-2 in Chapter 2), represents a peak potential stress of -181.3 psi at the inner concrete shell surface. As shown in the diagram above, the radial compressive stress decreases through the wall thickness. This stress is considered to be insignificant.

Vertical membrane and bending stress varies through the concrete wall as shown in the diagram.



A linear equation describes the stress as a function of wall thickness.

σ_v = 46.98 t - 525.4, where t = r - 43 and r is the radius from the centerline of the storage cask to the external surface of the concrete. Substituting for t:

= 46.98r - 2,545.54

Integration of vertical tensile stress over the area in the plane r- θ gives the tensile loads acting in the vertical direction.

$$F_v' = \int \sigma_v \; da = \int \int (46.98\,r - 2{,}545.54)\, r \, dr \, d\theta$$

$$= 863{,}706 \text{ lbf}$$

56 outer vertical reinforcing bars are equally spaced at a 60.63 in. radius, which is close to the 60.73 in. radius tensile load center. The maximum tensile load applied to the reinforcing bar is:

$F_{v\,applied} = F_v' / 56 = 15{,}423$ lbf per vertical reinforcing bar.

Using a 1.275 load factor for normal operating loads:

$1.275\, F_{v\,applied} = 1.275(15{,}423) = 19{,}664$ lbf

Calculating the allowable load for the reinforcing bar:

$$\sigma_{\theta\ allowable} = U_c = \phi\ S_y = 0.90\ (60) = 54\ \text{ksi}$$

where:

S_y = Reinforcing bar Yield Strength = 60 ksi

ϕ = 0.9 for axial and bending tension loading (strength reduction factor in accordance with ACI 349-85, Section 9.2)

The reinforcing bar tensile load capacity is:

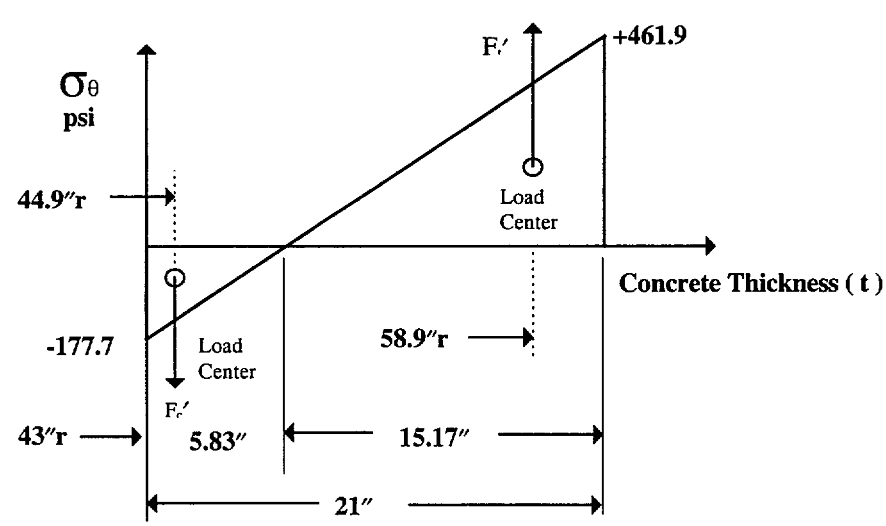
$$F_\theta = (\sigma_{\theta\ allowable})(A_{\#6}) = (54{,}000)\ (.44) = 23{,}760\ \text{lbs}$$

where, $A_{\#6}$ = #6 reinforcing bar area = 0.44 in.^{2}

The calculated load of 19,664 lbs is less than the allowable reinforcing bar load of 23,760 pounds. Therefore, the design is adequate with a margin of safety equal to:

$$M.S. \quad = \quad \frac{23{,}760}{19{,}664} - 1 = +\ 0.21$$

The circumferential membrane and bending stress varies through the concrete wall. The maximum values occur at 92.6 in. from the concrete cask lower surface as shown in the following diagram.



A linear equation describes the stress as a function of wall thickness:

$$\sigma_\theta = 30.457\,t - 177.7$$

Integration of circumferential stress over the area in plane r-y gives the load acting in the circumferential (θ) direction. Integration of this stress over the concrete wall thickness provides the distributed load per unit height.

Circumferential tensile loads are found by integrating the stress function (using integration limits for wall thickness of 5.83 to 21 in).

$$\begin{aligned} F_t' &= \int \sigma_\theta \;\; dt = \int (30.457\,t - 177.7)\,dt \\ &= (15.23\,t^2 - 177.7t)\Big|_{t=5.83}^{t=21} \\ &= 3{,}503.5 \text{ lbs/in} \end{aligned}$$

Outer hoop reinforcing bars are spaced on 4-in. centers at a 60.63 in. radius. The maximum circumferential tensile load acting on an outer hoop reinforcing bar in this spacing is:

$$F_{t\theta\,\text{applied}} = F_t' \times 4 = 14{,}014 \text{ lbs}$$

Using a 1.275 load factor for normal operating loads

$$1.275\,F_{t\theta\,\text{applied}} = 1.275(14{,}014) = 17{,}868 \text{ lbf}$$

The calculated load of 17,868 lbf is less than the reinforcing bar allowable of 23,760 lbf. Therefore, the design is adequate with a margin of safety equal to:

$$\text{M.S.} \quad = \quad \frac{23{,}760}{17{,}868} - 1 = +\,0.33$$

Figure 3.4.4.2-1 Concrete Cask Axisymmetric Thermal Stress Model

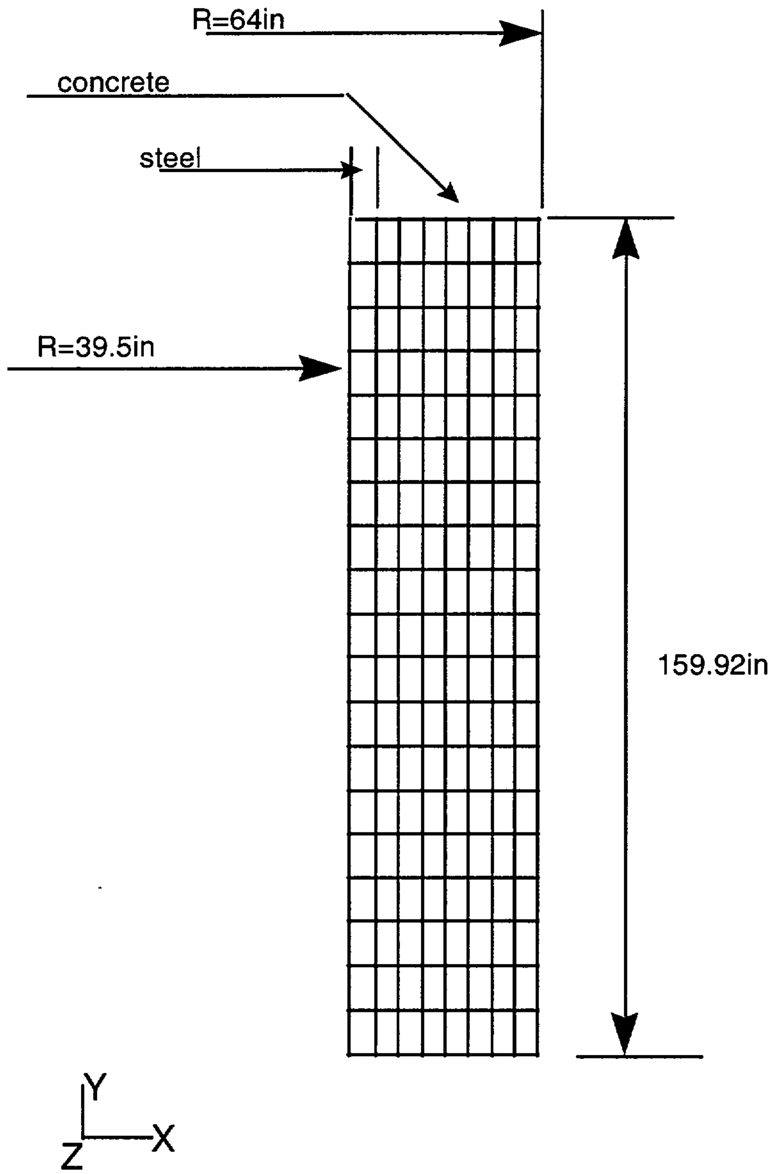


Table 3.4.4.2-1 Stress Summary for Concrete Cask Load Combinations

Load Comb[1]	Stress Direction	Stress[2] (psi)							
		Dead	Live	Wind[3]	Thermal[4]	Seismic[5]	Tornado[6]	Flood[7]	Total
CONCRETE OUTSIDE SURFACE:									
1	Vertical	-30.0	-32.6	---	---	---	---	---	-45.2
2	Vertical	-22.5	-24.5	---	---	---	---	---	-47.0
3	Vertical	-22.5	-24.5	-14.7	---	---	---	---	-61.7
4	Vertical	-21.4	-19.2	---	---	---	---	---	-40.6
5	Vertical	-21.4	-19.2	---	---	-42.9	---	---	-83.5
7	Vertical	-21.4	-19.2	---	---	---	---	-10.6	-51.2
8	Vertical	-21.4	-19.2	---	---	---	-11.5	---	-52.1
CONCRETE INSIDE SURFACE:									
1	Vertical	-30.0	-32.6	---	---	---	---	---	-62.2
	Circumferential	---	---	---	---	---	---	---	---
2	Vertical	-22.5	-24.5	---	-669.9	---	---	---	-716.9
	Circumferential	---	---	---	-226.6	---	---	---	-226.6
3	Vertical	-22.5	-24.5	-9.9	-669.9	---	---	---	-726.8
	Circumferential	---	---	---	-226.6	---	---	---	-226.6
4	Vertical	-21.4	-19.2	---	-660.1	---	---	---	-700.7
	Circumferential	---	---	---	-127.5	---	---	---	-127.5
5	Vertical	-21.4	-19.2	---	-525.4	-31.2	---	---	-597.2
	Circumferential	---	---	---	-177.7	---	---	---	-177.7
7	Vertical	-21.4	-19.2	---	-525.4	---	---	-7.1	-573.1
	Circumferential	---	---	---	-177.7	---	---	---	-177.7
8	Vertical	-21.4	-19.2	---	-525.4	---	-7.7	---	-573.7
	Circumferential	---	---	---	-177.7	---	---	---	-177.7

[1] Load Combinations are defined in Table 2.2-2. See Section 11.2.11 and 11.2.12 for Evaluations of Drop/Impact Conditions for Load combination No. 6.

[2] Positive stress values indicate tensile stresses and negative values indicate compressive stresses.

[3] Stress results from Section 11.2.13 (Tornado) are conservatively used with a load factor of 1.275.

[4] Tensile stresses (at concrete outside surface) are taken by the steel reinforcing bars and therefore are not shown in this Table. Stress Results for T_a (Load Comb. #4) are obtained from Section 11.2.10.

[5] Stress results are obtained from Section 11.2.2.

[6] Stress results are obtained from Section 11.2.13 (Tornado Wind).

[7] Stress results are obtained from Section 11.2.6.

Table 3.4.4.2-2 Maximum Concrete Stress and Reinforcing Bar Forces

	Calculated	Allowable[1]	Margin of Safety
Concrete	727 psi	2,800 psi	+2.85
Reinforcing Bar			
Normal - vertical	19,664 lbs	23,760 lbs	+0.21
- hoop	17,868 lbs	23,760 lbs	+0.33
Accident[2] - vertical	19,380 lbs	23,760 lbs	+0.22
- hoop	23,196 lbs	23,760 lbs	+0.02

1 Allowable stress for concrete is (0.7)(4,000 psi) = 2,800 psi, where 0.7 is the strength reduction factor per ACI 349-85, Section 9.3; 4,000 psi is the concrete strength.
Allowable for Reinforcing Bar is determined based on No. 6 Reinforcing Bar as shown in the calculation in this Section.

2 Results are obtained from Section 11.2.10.

3.4.5 Cold

Severe cold environments are analyzed and reported in Section 11.1.4. As shown in that section, the temperature of the structures with a full heat load will not fall to levels where brittle fracture would become an issue. Furthermore, an analysis has been performed for the cask in severe cold conditions after 50 years of storage. The required material toughness is 12.6 ft-lbs at -30°F. For conservatism 15 ft-lbs at -50°F is stated in the fabrication specification so that the NAC-MPC system can be handled, even during extreme temperatures.

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3.5 Fuel Rods

The NAC-MPC system is designed to limit fuel cladding temperatures to levels below those where zircaloy degradation is expected to lead to fuel clad failure. As shown in Chapter 4, fuel cladding temperature limits have been established to be 380°C for 5-year cooled fuel and 340°C for 10-year cooled fuel for normal conditions of storage and 570°C for short term off-normal and accident conditions. As shown in Table 4.1-3, the calculated maximum fuel cladding temperatures are well below the temperature limits for all design conditions of storage.

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3.6 Canister Closure Weld Evaluation – Normal Conditions

3.6.1 Stress Evaluation for the Canister Closure Weld

The closure weld for the canister is a partial penetration weld with a thickness of 0.9 inches. The evaluation of this weld, in accordance with NRC guidance, is to incorporate two separate weld stress reduction factors: a 0.8 factor per NRC ISG-4, Item 5, and an additional 0.8 factor to provide conservative consideration of the weld configuration. These two stress reduction factors are incorporated by applying a factor of 0.64 (0.8 x 0.8) to the stress allowable for this weld.

The stresses for the canister are evaluated using sectional stresses as permitted by Subsection NB. The location of the section for the canister weld evaluation is shown in Figure 3.4.4.1-4 and corresponds to Section 12. The governing P_m, $P_m + P_b$, and P + Q stress intensity for Section 12 and the associated allowables are listed in Table 3.4.4.1-8, Table 3.4.4.1-9, and Table 3.4.4.1-10, respectively. The factored allowables, incorporating a 0.64 stress reduction factor, and the resulting controlling Margin of Safeties are:

Stress Type	Analysis Stress Intensity (ksi)	0.64 x Allowable Stress (ksi)	Margin of Safety
P_m	1.19	10.69	7.98
$P_m + P_b$	1.66	16.03	8.66
P + Q	6.99	32.06	3.59

This confirms that the canister closure weld is acceptable for normal operation conditions.

3.6.2 Critical Flaw Size for the Canister Closure Weld

The closure weld for the canister is comprised of multiple weld beads using a compatible weld material for Type 304L stainless steel. An allowable (critical) flaw evaluation has been performed to determine the critical flaw size in the weld region. The result of the flaw evaluation is used to define the minimum flaw size, which must be identifiable in the nondestructive examination of the weld. Due to the inherent toughness associated with Type 304L stainless steel, a limit load analysis is used in conjunction with a J-integral/tearing modulus approach.

The safety margins used in this evaluation correspond to the stress limits contained in Section XI of the ASME Code.

One of the stress components used in the evaluation for the critical flaw size is the radial stress component in the weld region of the structural lid. For the normal operation condition, in accordance with ASME Code Section XI, a safety factor of 3 is required. For the purpose of identifying the stress for the flaw evaluation, the weld region corresponds to Section 12 Figure 3.4.4.1-4 is considered. The radial stress corresponds to SX in Tables 3.4.4.1-1 through 3.4.4.1-10 for the normal operating condition. The maximum reported radial tensile stress is 0.7 ksi. To perform the flaw evaluation, a 10 ksi stress is conservatively used, resulting in a significantly larger safety factor than the required safety factor of 3. Using 10 ksi as the basis for the evaluation, the critical flaw size is 0.52 inch for a flaw that extends 360 degrees around the circumference of the canister. Stress components for the circumferential and axial directions are also reported in the above tables, which would be associated with flaws oriented in the radial or horizontal directions, respectively. The maximum stress reported for these components is 1.8 ksi, which is also enveloped by the stress value of 10 ksi for radial stresses. The 360-degree flaw employed for the circumferential direction is considered to be bounding with respect to any partial flaw in the weld, which could occur in the radial and horizontal directions. Therefore, using a minimum detectable flaw size of 0.375 inch is acceptable, since it is less than the very conservatively determined 0.52-inch critical flaw size.

3.7 References

ACI-349-85, Code Requirements for Nuclear Safety Related Concrete Structures, American Concrete Institute, March 1986.

ANSI N 14.6-1993, American National Standard for Radioactive Material - Special Lifting Devices for Shipping Containers Weighing 10,000 Pounds (4,500 kg) or More, American National Standards Institute, Inc., 1993.

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Manual of Steel Construction, Allowable Stress Design, Ninth Edition, American Institute of Steel Construction, Inc., 1 East Wacker Drive, Chicago, Illinois, 1991.

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TRW, Nelson Division, "Embedment Properties of Headed Studs," Design Data 10, 1975.

Young, W. C., Roark's formulas for Stress & Strain, 6th Edition, McGraw-Hill Book Company, New York, 1989.

3.8 Coating Specifications

This Section presents the technical data sheets for Keeler & Long E-Series epoxy enamel and Ameron PSX 738. These coatings are applied to protect exposed carbon steel surfaces of the transfer cask and vertical concrete cask.

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3.8.1 Keeler and Long E-Series Epoxy Enamel

March, 1995 SSU-1

KEELER & LONG INC.

HEADQUARTERS:
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856 Echo Lake Road
Watertown, CT 06795
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PROTECTIVE COATING SYSTEMS FOR NUCLEAR POWER PLANTS

INTRODUCTION

In the 1960's Keeler & Long made the commitment to develop Protective Coating Systems for Nuclear Power Plants. Coating Systems were developed and qualified in accordance with accepted standards, with emphasis upon their usage and specification for NEW construction projects. These systems were applied directly to either concrete or carbon steel substrates utilizing ideal surface preparation.

Presently, there is a necessity to apply these same coating systems or newly formulated systems over the original systems or over substrates which cannot be ideally prepared. Several years ago, Keeler & Long initiated a test program in order to test and qualify systems in conjunction with competitors products and/or with methods of preparation which are considered less than ideal. This test program provides OPERATING Nuclear Plants with qualified methods of preparation and a variety of qualified mixed coating systems.

HISTORY

In 1967, we embarked upon a testing program in order to comply with standards being prepared by the experts in the field and under the jurisdiction of The American National Standards Institute (ANSI). Earlier testing had involved research in order to determine the radiation tolerance and the decontamination properties of a variety of generic coating types including zinc rich, alkyds, chlorinated rubbers, vinyls, latex emulsions, and epoxies. This testing was conducted by various independent laboratories, such as Oak Ridge National Laboratory, Idaho Nuclear, and The Western New York Nuclear Research Center. It was concluded from these tests that almost any generic coating type would produce satisfactory radiation resistance and decontaminability.

Upon completion of the first ANSI Standards, however, it became evident that only Epoxy Coatings would meet the specific minimum acceptance criteria set forth in these standards. The single most important change from the earlier testing was the inclusion of a test which simulates the operation of the emergency core cooling system. This test is referred to as the Loss of Coolant Accident (LOCA) or the Design Basis Accident Condition (DBA). The test involves a high pressure, high temperature, alkaline, immersion environment.

Simultaneous with the preparation of these standards, we prepared to test Epoxy Systems in order to comply with the requirements. First hand knowledge of these standards was available since our personnel assisted in the development of these documents. Equipment was designed and built by our laboratory in order to conduct in-house DBA tests. The required physical and chemical tests were either conducted by us or by universities through research grants.

In 1972, the testing program was taken a step further in order to establish more credibility. The Franklin Institute of Philadelphia constructed an apparatus in order to simulate various Design Basis Accident Conditions and we prepared blocks and panels for an independent evaluation. The test results were among the "First" from an independent source, and these tests substantiated more than two years of in-house testing.

The Franklin Institute tests, along with our in-house testing program, were used as a basis for qualification until 1976. During this period also the following ANSI standards were revised and/or developed:

> ANSI N5.9-1967 "Protective Coatings (Paints) for the Nuclear Industry" (Rev. ANSI N512-1974)
>
> ANSI N101.2-1972 "Protective Coatings (Paints) for Light Water Nuclear Reactor Containment Facilities"
>
> ANSI N101.4-1972 "Quality Assurance for Protective Coatings Applied to Nuclear Facilities"

Simultaneously, we developed a written Quality Assurance Program in compliance with ANSI N101.4 - 1972, Appendix B 10CFR50 of the Federal Register, and ANSI N45.2-1971 "Quality Assurance Program Requirements For Nuclear Power Plants".

In 1976, Oak Ridge national Laboratory (ORNL) established a testing program in order to conduct Radiation, Decontamination, and DBA tests under one roof. Keeler & Long, under contract with ORNL, conducted a series of tests in compliance with the parameters established by a major engineering firm and the ANSI standards. These tests, and similar series of tests conducted two years later in 1978, became the basis for the qualification of several of our concrete and carbon steel coating systems. From 1978 to the present day we have continued to qualify through ORNL and several other independent testing agencies any modifications to existing formulas and any changes in surface preparation or application requirements. We have also maintained an in-house testing program used to screen new products as well as modifications of existing systems. Furthermore, progress has continued in the revision of the ANSI standards during this time frame. Revision of these documents is presently under the jurisdiction of the American Society for Testing and Materials (ASTM) as outlined in D3842-80 "Standard Guide for Selection of Test Methods for Coatings Used In Light-Water Nuclear Power Plants".

The future dictates significantly less construction of new Nuclear Plants and much more emphasis upon the repair and maintenance of existing facilities. Our commitment remains the same as it was in 1965; that is, to meet the coating requirements of Nuclear Power Plants.

NUCLEAR COATINGS

SSU-1

Level One Coating Systems

The following Coating Systems are qualified for Coating Service Level One of a Nuclear Power Plant. "Coating Service Level One pertains to those systems applied to structures, systems and other safety related components which are essential to the prevention of, or the mitigation of the consequences of postulated accidents that could cause undue risk to the health and safety of the public."

SYSTEM IDENTIFICATION	COATING SYSTEMS	DRY FILM THICKNESS RANGE
CARBON STEEL COATING SYSTEMS		
System S-1		
Primer	No. 6548/7107 EPOXY WHITE PRIMER	3.0 - 14.0 mils DFT
Finish	No. E-1 SERIES EPOXY ENAMEL	2.5 - 6.0 mils DFT
System S-10		
Primer	No. 6548/7107 EPOXY WHITE PRIMER	5.0 - 12.0 mils DFT
Finish	No. D-1 SERIES EPOXY HI-BUILD ENAMEL	3.0 - 6.0 mils DFT
System S-11		
Primer/Finish	No. 6548/7107 EPOXY WHITE PRIMER	8.0 - 18.0 mils DFT
System S-12		
Primer/Finish	No. 4500 EPOXY SELF-PRIMING SURFACING ENAMEL	5.0 - 18.0 mils DFT
System S-14 (FLOORS ONLY)		
Finish	No. 5000 EPOXY SELF-LEVELING FLOOR COATING	10.0 - 25.0 mils DFT
System S-15		
Primer	No. 6548/7107 EPOXY WHITE PRIMER	2.5 - 6.0 mils DFT
Finish	No. 9600 N KEELOCK	5.0 - 8.0 mils DFT
CONCRETE COATING SYSTEMS		
System KL-2		
Curing Compound/Sealer	No. 4129 EPOXY CLEAR CURING COMPOUND	0.5 - 1.75 mils DFT
Surfacer	No. 6548-S EPOXY SURFACER	Flush - 50.0 mils DFT
Finish	No. E-1 SERIES EPOXY ENAMEL	2.5 - 6.0 mils DFT
System KL-8		
Curing Compound/Sealer	No. 4129 EPOXY CLEAR CURING COMPOUND	0.5 - 1.75 mils DFT
Surfacer	No. 6548-S EPOXY SURFACER	Flush - 50.0 mils DFT
Finish	No. D-1 SERIES EPOXY HI-BUILD ENAMEL	4.0 - 8.0 mils DFT
System KL-9		
Curing Compound/Sealer	No. 4129 EPOXY CLEAR CURING COMPOUND	0.5 - 1.75 mils DFT
Surfacer	No. 6548/7107 EPOXY WHITE PRIMER	5.0 - 10.0 mils DFT
Finish	No. D-1 SERIES EPOXY HI-BUILD ENAMEL	3.0 - 8.0 mils DFT
System KL-10		
Curing Compound/Sealer	No. 4129 EPOXY CLEAR CURING COMPOUND	0.5 - 1.75 mils DFT
Surfacer	No. 4000 EPOXY SURFACER	Flush - 50.0 mils DFT
Finish	No. D-1 SERIES EPOXY HI-BUILD ENAMEL	3.0 - 6.0 mils DFT
System KL-12		
Curing Compound/Sealer	No. 4129 EPOXY CLEAR CURING COMPOUND	0.5 - 1.75 mils DFT
Surfacer/Finish	No. 4500 EPOXY SELF-PRIMING SURFACING ENAMEL	10.0 - 50.0 mils DFT
System KL-14 (FLOORS ONLY)		
Primer/Sealer	No. 6129 EPOXY CLEAR PRIMER/SEALER	1.5 - 2.5 mils DFT
Finish	No. 5000 EPOXY SELF-LEVELING FLOOR COATING	35.0 - 50.0 mils DFT

SUMMARY OF QUALIFICATION TEST RESULTS

KEELER & LONG maintains a complete file of Nuclear Test Reports which substantiate the specification of the carbon steel and concrete coating systems listed in this bulletin. This file was initiated in the early 1970's and provides complete qualification in accordance with ANSI Standards N512 and N101.2. Results for radiation tolerance, decontamination, and the Design Basis Accident Condition are reported as performed by independent Laboratories. Also reported are the chemical and physical tests which were conducted by the Keeler & Long Laboratory in compliance with the ANSI Standards.

TEST REPORT REFERENCE

K&L COATING SYSTEM	SUBSTRATE	KEELER & LONG TEST REPORT NO. 76-0728-1	78-0810-1	85-0404	85-0524	90-0227	93-0818	93-0601
S-1	Steel	*	*					
S-10	Steel		*					
S-11	Steel		*					
S-12	Steel			*				
S-14	Steel					*		
S-15	Steel						*	
KL-2	Concrete	*	*					
KL-8	Concrete	*						
KL-9	Concrete	*	*					
KL-10	Concrete		*					
KL-12	Concrete				*			
KL-14	Concrete					*		*

E.340

HEADQUARTERS:
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EPOXY ENAMEL
E-SERIES

GENERIC TYPE: POLYAMIDE EPOXY

PRODUCT DESCRIPTION: A two component, polyamide epoxy enamel formulated to provide excellent chemical resistance, as well as being extremely resistant to abrasion and direct impact, for interior exposures.

RECOMMENDED USES: As a topcoat for concrete and steel surfaces subject to radiation, decontamination, and loss-of-coolant accidents in Coating Service Level I Areas of nuclear power plants.

NOT RECOMMENDED FOR: Areas other than the above, as the J-SERIES can be utilized in Coating Service Level II and III Areas, as well as Balance of Plant, of nuclear power plants, with attendant cost savings.

COMPATIBLE UNDERCOATS: Epoxy White Primer
Epoxy Surfacer

PRODUCT CHARACTERISTICS:

Solids by Volume:	53% ± 3%
Solids by Weight:	66% ± 3%
Recommended Dry Film Thickness:	2.0 - 2.5 mils
Theoretical Coverage:	425 Sq. Ft./Gallon @ 2.0 mils DFT
Finish:	Full Gloss (E-1), Semi-Gloss (E-2)
Available Colors:	White, light tints, and dark red
Drying Time @ 72°F	
To Touch:	4 Hours
To Handle:	8 Hours
To Recoat:	48 Hours
VOC Content:	3.4 Pounds/Gallon 407 Grams/Liter

June, 1994

TECHNICAL BULLETIN

E-SERIES

E 340

TECHNICAL DATA

PHYSICAL DATA:	Weight per gallon:	10.2 ± 0.5 (pounds)
	Flash Point (Pensky-Martens):	85°F± 2°
	Shelf Life:	1 Year
	Pot Life @ 72°F:	8 Hours
	Temperature Resistance:	350°F
	Viscosity @ 77°F:	85 ± 5 (Krebs Units)
	Gloss (60° meter):	95 ± 5 (E-1)
	Storage Temperature:	55 - 95°F
	Mixing Ratio (Approx. by Volume):	4:1
APPLICATION DATA:	Application Procedure Guide:	APG-2
	Wet Film Thickness Range:	4.0 - 5.0 mils
	Dry Film Thickness Range:	2.0 - 2.5 mils
	Temperature Range:	55 - 120°F
	Relative Humidity:	80% Maximum
	Substrate Temperature:	Dew Point + 5°F
	Minimum Surface Preparation:	Primed
	Induction Time @ 72°F:	1 Hour
	Recommended Solvent	
	@ 50 - 85°F:	No. 4093
	@ 86 - 120°F:	No. 2200

Application Methods

Air Spray	
Tip Size:	.055"
Pressure:	30 - 60 PSIG
Thin:	1.0 - 2.0 Pts/Gal
Airless Spray	
Tip Size:	.011" - .017"
Pressure:	2500 - 3000 PSIG
Thin:	0.5 - 1.5 Pts/Gal
Brush or Roller	
Thin:	1.0 - 2.0 Pts/Gal

KEELER & LONG INC.

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KOLORS

SSPC SUSTAINING MEMBER

This information is presented as accurate and correct, in good faith, to assist the user in specification and application. No warranty is expressed or implied. No liability is assumed. Product specifications are subject to change without notice. Data listed above is for white or base color of the product. Data for other colors may differ.

3.8.2 Ameron PSX 738 Siloxane Coating

Ameron®

PSX® 738

Engineered Siloxane® coating

Patent No. 4,113,665

PSX Advantage: PSX 738 is a patented siloxane product that can withstand twice as much heat as conventional heat-resistant coatings, without the need for heat curing before service.

Product Data

- Protects stainless steel from chloride attack under insulation
- Heat resistance to 2000°F
- Self-priming, single-coat efficiency—significantly lower applied cost
- Resistant to severe acid conditions and moisture
- Chemical resistant
- Excellent adhesion
- Suitable to protect steel under thermal insulation at high temperatures
- Cures at room temperature, no bake required before service
- Applied direct to metal
- High solids and low VOC

PSX 738 is resistant to a wide range of mineral and organic acids, including glacial acetic acid. (See Chemical Resistance Guide - below).

Typical Uses

- Power plants
- Exhaust ducts and stacks
- Refineries
- Exteriors of reactors
- Chemical facilities
- Under insulation, on equipment and pipe exteriors

PSX 738 Chemical Resistance Guide

PSX 738 may be exposed to the following chemicals under splash and spillage conditions:

Chemical	Temperature
Salt water (3%)	up to 200°F
DI Water	up to 200°F
Sodium silicate	Ambient
Ethylene glycol	Ambient
Acetone	Ambient
Methylene chloride	Ambient
Xylene	Ambient
Mineral spirits	Ambient
JP-4 fuel	up to 140°F
Petroleum ether	Ambient
Kerosene	up to 140°F
Transmission fluid	Ambient
Acetic acid (all concentrations)	Ambient
Tall oil fatty acid	up to 120°F
Sulfuric acid (50%)	Ambient
Nitric acid (5%)	Ambient
Hydrochloric acid(20%)	Ambient

Note: Not recommended for caustic solutions. PSX 738 is not recommended for the lining of mineral acid storage tanks.

Physical Data

Finish	Flat	
Color*	Deep gray (approx ANSI-33), gray (approx GR-3)	
Components	2	
Curing mechanism	Chemical reaction and solvent release	
Volume solids (ASTM D2697 modified)	84% ± 3%	
Dry film thickness per coat	4 - 6 mils (100 - 150 microns)	
Refer to maximum DFT table for service temperature limitations.		
Coats	1- for high heat, over Dimetcote® and hot steel 2- highly corrosive service	
Theoretical coverage	ft²/gal	m²/L
1 mil (25 microns)	1347	33.0
10 mils (250 microns)	135	3.3
VOC	lb/gal	g/L
mixed	0.8	96
mixed/thinned (½ pt/gal)	1.2	144
Temperature resistance, dry	°F	°C
continuous	2000	1093
over Dimetcote 9 or 21-9	750	400
Flash point (SETA)	°F	°C
liquid	55	13
Amercoat 935	208	98
Amercoat 101	145	63
Amercoat 15	106	41
Amercoat 12	0	-18

Application Data

Applied over	Prepared or primed steel, stainless steel types 304 & 316, concrete or Dimetcote	
Surface preparation		
steel or stainless steel types 304 & 316	Abrasive blast SSPC-SP10	
concrete	ASTM D4258	
Dimetcote 9 or 21-9	Clean, dry surface	
Method	Conventional spray	
Mixing ratio (by weight)	1 part liquid to 1.9 parts powder	
Pot life @ 70°F and 50% R.H.	5 hours	
Environmental conditions Temperature	°F	°C
air	45 to 120	7 to 49
surface	See thinner recommendations	
Relative humidity during application and initial drying		
minimum	40%	
maximum	Avoid condensation	

Surface temperatures must be at least 5°F (3°C) above dew point to prevent condensation during application and initial dry through.

**Note- color will lighten at high temperatures*

Formerly Amercoat 738

Surface Preparation

Coating performance is, in general, proportional to the degree of surface preparation. Prior to coating all surfaces must be clean, dry, undamaged and free of all contaminants, including salt deposits.

Application Data Summary

See Application Instructions for complete information on surface preparation, equipment, environmental conditions and application procedures. For conditions outside the requirements or limitations described, contact your Ameron representative.

Maximum DFT (mils)

	Thinner	No. of Coats	Service Temperature °F/°C 2000/1093	1500/816	1000/538	750/399	500/260
Steel	15/101	1	8	10	10	10	15
	15/101	2	NR	10	10	14	20
	935	1	NR	8	8	12	15
Dimetcote	15/101	1	—	—	—	10	15
	935	NR	NR	NR	NR	NR	NR

NR= Not recommended

Important: Exceeding maximum DFT may result in PSX 738 surface cracking.

Safety Precautions

Read each component's material safety data sheet before use. Mixed material has hazards of each component. Safety precautions must be strictly followed during storage, handling and use.

This product is for professional use only. Not for residential use in California.

Warranty

Ameron warrants its products to be free from defects in material and workmanship. Ameron's sole obligation and Buyer's exclusive remedy in connection with the products shall be limited, at Ameron's option, to either replacement of products not conforming to this Warranty or credit to Buyer's account in the invoiced amount of the nonconforming products. Any claim under this Warranty must be made by Buyer to Ameron in writing within five (5) days of Buyer's discovery of the claimed defect, but in no event later than the expiration of the applicable shelf life, or one year from the delivery date, whichever is earlier. Buyer's failure to notify Ameron of such nonconformance as required herein shall bar Buyer from recovery under this Warranty.

Ameron makes no other warranties concerning the product. No other warranties, whether express, implied, or statutory, such as warranties of merchantability or fitness for a particular purpose, shall apply. In no event shall Ameron be liable for consequential or incidental damages.

Any recommendation or suggestion relating to the use of the products made by Ameron, whether in its technical literature, or in response to specific inquiry, or otherwise, is based on data believed to be reliable; however, the products and information are intended for use by Buyers having requisite skill and know-how in the industry, and therefore it is for Buyer to satisfy itself of the suitability of the products for its own particular use and it shall be deemed that Buyer has done so, at its sole discretion and risk. Variation in environment, changes in procedures of use, or extrapolation of data may cause unsatisfactory results.

Limitation of Liability

Ameron's liability on any claim of any kind, including claims based upon Ameron's negligence or strict liability, for any loss or damage arising out of, connected with, or resulting from the use of the products, shall in no case exceed the purchase price allocable to the products or part thereof which give rise to the claim. **In no event shall Ameron be liable for consequential or incidental damages.**

Drying time** (ASTM D1640)(hours)

°F/°C	120/49	90/32	70/21	50/10	40/4
Touch @ 50% R.H.					
unthinned	1/2	1	1 1/2	2	3
thinned w/15 or 101***	1/2	2	3	4	6
Through @ 50% R.H.					
unthinned	14	20	28	48	72
thinned w/15 or 101***	14	26	36	60	80
Through @ 75% R.H.					
unthinned	7	10	14	24	36
thinned w/15 or 101***	7	15	18	30	40
Recoat @ 50% R.H.					
unthinned	1	2	3	5	7
thinned***	2	4	6	10	14
Recoat @ 75% R.H.					
unthinned	1	1	2	3	4
thinned***	1 1/2	3	5	8	10
Cure** (days)					
Service temp. up to 1000°F – same as Dry Through time					
Service temp. above 1000°F (thinned or unthinned)					
@ 50% R.H.	4	8	13	30	50
@ 75% R.H.	2 1/2	4	7	19	32
Chemical resistance (days)					
@50% R.H.					
unthinned	6	10	15	40	80
thinned***	10	15	25	60	NR
@ 75% R.H.					
unthinned	3	6	10	21	40
thinned***	6	10	15	36	72

NR = Not Recommended

***Note – at 15 mils total DFT Drying and Cure times will increase by 30%.*

****Note – Thinned material at 2 coats will increase cure times.*

Thinners

Amercoat 15 (up to 100°F surface temperature)
Amercoat 101 (up to 140°F surface temperature)
Amercoat 935 (140° - 200°F surface temperature)

DO NOT use Amercoat 935 below 140°F surface temperatures.

Equipment cleaner — Thinner or Amercoat 12

Shipping Data

Packaging units	1-gal	5-gal
Shipping weight (approx)	lb	kg
1-gal unit		
liquid	6.1	2.8
powder	11.1	5.1
5-gal unit		
liquid	30.4	13.8
powder	52.9	24.1

Shelf life when stored indoors at 40 to 100°F (4 to 38°C)
liquid — 8 months from manufacture date
powder — 2 years from shipment date

Numerical values are subject to normal manufacturing tolerances, color and testing variances. Allow for application losses and surface irregularities.

The mixed product is nonphotochemically reactive as defined by the South Coast Air Quality Management District's Rule 102 or equivalent regulations.

Ameron.

Protective Coatings Systems

201 North Berry Street, Brea, California 92622-1020 • J. F. Kennedylaan 7, 4191 MZ Geldermalsen, The Netherlands

738 PDS

Ameron®

PSX® 738

Engineered Siloxane® coating

Patent No. 4,113,665

Application Instructions

Refer to the Product Data sheet for properties and uses.

Adhere to all application instructions, precautions, conditions, and limitations to obtain the maximum performance. For conditions outside the requirements or limitations described, contact your Ameron representative.

Surface Preparation

Coating performance is, in general, proportional to the degree of surface preparation. Prior to coating all surfaces must be clean, dry, undamaged and free of all contaminants, including salt deposits.

Steel or Stainless Steel Types 304 & 316 – Dry abrasive blast according to SSPC-SP10.

Concrete – Clean according to ASTM D4258.

Dimetcote® 9 or 21-9 – Clean dry surface. See Dimetcote 9 or 21-9 Application Instructions for complete information and safety precautions.

Application Equipment

The following is a guide: suitable equipment from other manufacturers may be used. Changes in pressure, hose and tip size may be needed for proper spray characteristics.

Conventional spray – Industrial equipment such as DeVilbiss MBC or JGA spray gun. An oil and moisture trap in the main air supply line are essential. Separate regulators for air and fluid pressure and mechanical pot agitator are recommended.

Environmental Conditions

	°F	°C
Temperature		
air	45 to 120	7 to 49
surface	See thinner recommendations	
Relative humidity during application and initial drying		
minimum	40%	
maximum	Avoid condensation	

Surface temperatures must be at least 5°F (3°C) above dew point to prevent condensation during application and initial dry through.

Application Procedure

1. Flush equipment with thinner or Amercoat 12 before use to remove moisture. Moisture can harden Amercoat 738 in spray equipment.
2. Slowly stir powder into liquid and continue to mix until powder is well dispersed. Do not reverse order.
3. Keep mixture in covered container until ready to use, as moisture from the atmosphere can cause skinning or gelling. Do not mix more material than can be used within the expected pot life, 5 hours at 70°F and 50% R.H.
4. Agitate slowly and continuously during spray application to maintain uniform suspension.
5. Apply a wet coat in multiple passes, if necessary, to obtain desired dry film thickness. **Maintain 18" maximum distance between spray gun and surface to avoid dry spray.**

Maximum DFT (mils)

	Thinner	No. of Coats	Service Temperature °F/°C 2000/1093	1500/816	1000/538	750/399	500/260
Steel	15/101	1	8	10	10	10	15
	15/101	2	NR	10	10	14	20
	935	1	NR	8	8	12	15
Dimetcote	15/101	1	—	—	—	10	15
	935	NR	NR	NR	NR	NR	NR

NR= Not recommended

Important: Exceeding maximum DFT may result in PSX 738 surface cracking.

Drying time* (ASTM D1640)(hours) °F/°C

	120/49	90/32	70/21	50/10	40/4
Touch @ 50% R.H.					
unthinned	1/2	1	1 1/2	2	3
thinned w/ 15 or 101**	1/2	2	3	4	6
Through @ 50% R.H.					
unthinned	14	20	28	48	72
thinned w/15 or 101**	14	26	36	60	80
Through @ 75% R.H.					
unthinned	7	10	14	24	36
thinned w/15 or 101**	7	15	18	30	4
Recoat @ 50% R.H.					
unthinned	1	2	3	5	7
thinned**	2	4	6	10	14
Recoat @ 75% R.H.					
unthinned	1	1	2	3	4
thinned**	1 1/2	3	5	8	10
Cure* (days)					
Service temp. up to 1000°F – same as Dry Through time					
Service temp. above 1000°F (thinned or unthinned)					
@ 50% R.H.	4	8	13	30	50
@ 75% R.H.	2 1/2	4	7	19	32
Chemical resistance (days)					
@50% R.H.					
unthinned	6	10	15	40	80
thinned**	10	15	25	60	NR
@ 75% R.H.					
unthinned	3	6	10	21	40
thinned**	6	10	15	36	72

NR = Not Recommended

**Note – at 15 mils total DFT Drying and Cure times will increase by 30%.*
***Note – Thinned material at 2 coats will increase cure times.*

6. Thin with recommended thinner as required for workability, see below:

Thinners
- Amercoat 15 (up to 100°F surface temperature)
- Amercoat 101 (up to 140°F surface temperature)
- Amercoat 935 (140° - 200°F surface temperature)

DO NOT use Amercoat 935 below 140°F surface temperatures.

7. Clean all equipment with thinner or Amercoat 12 cleaner immediately after use.

Formerly Amercoat 738

Safety Precautions

Read each component's material safety data sheet before use. Mixed material has hazards of each component. Safety precautions must be strictly followed during storing, handling and use.

CAUTION – Improper use and handling of this product can be hazardous to health and cause fire or explosion.

Do not use this product without first taking all appropriate safety measures to prevent property damage and injuries. These measures may include, without limitation: implementation of proper ventilation, use of proper lamps, wearing of proper protective clothing and masks, tenting and proper separation of application areas. Consult your supervisor. Proper ventilation and protective measures must be provided during application and drying to keep spray mists and vapor concentrations within safe limits and to protect against toxic hazards. Necessary safety equipment must be used and ventilation requirements carefully observed, especially in confined or enclosed spaces, such as tank interiors and buildings.

This product is to be used by those knowledgeable about proper application methods. Ameron makes no recommendation about the types of safety measures that may need to be adopted because these depend on application environment and space, of which Ameron is unaware and over which it has no control.

If you do not fully understand these warnings and instructions or if you cannot strictly comply with them, do not use the product.

Note: Consult Code of Federal Regulations Title 29, Labor, parts 1910 and 1915 concerning occupational safety and health standards and regulations, as well as any other applicable federal, state and local regulations on safe practices in coating operations.

This product is for professional use only. Not for residential use in California.

Warranty

Ameron warrants its products to be free from defects in material and workmanship. Ameron's sole obligation and Buyer's exclusive remedy in connection with the products shall be limited, at Ameron's option, to either replacement of products not conforming to this Warranty or credit to Buyer's account in the invoiced amount of the nonconforming products. Any claim under this Warranty must be made by Buyer to Ameron in writing within five (5) days of Buyer's discovery of the claimed defect, but in no event later than the expiration of the applicable shelf life, or one year from the delivery date, whichever is earlier. Buyer's failure to notify Ameron of such nonconformance as required herein shall bar Buyer from recovery under this Warranty.

Ameron makes no other warranties concerning the product. No other warranties, whether express, implied, or statutory, such as warranties of merchantability or fitness for a particular purpose, shall apply. In no event shall Ameron be liable for consequential or incidental damages.

Any recommendation or suggestion relating to the use of the products made by Ameron, whether in its technical literature, or in response to specific inquiry, or otherwise, is based on data believed to be reliable; however, the products and information are intended for use by Buyers having requisite skill and know-how in the industry, and therefore it is for Buyer to satisfy itself of the suitability of the products for its own particular use and it shall be deemed that Buyer has done so, at its sole discretion and risk. Variation in environment, changes in procedures of use, or extrapolation of data may cause unsatisfactory results.

Limitation of Liability

Ameron's liability on any claim of any kind, including claims based upon Ameron's negligence or strict liability, for any loss or damage arising out of, connected with, or resulting from the use of the products, shall in no case exceed the purchase price allocable to the products or part thereof which give rise to the claim. **In no event shall Ameron be liable for consequential or incidental damages.**

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Protective Coatings
Systems

738 AJ

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4.0 THERMAL EVALUATION

4.1 Discussion

This section presents the thermal analysis of the NAC-MPC system for normal conditions of storage. The significant thermal design feature of the NAC-MPC system is the passive convective air flow up along the side of the canister. Cool (ambient) air enters at the bottom of the storage cask through four inlet vents. Heated air exits through the four outlets at the top of the storage cask. Radiant heat transfer also occurs from the canister shell to the concrete cask liner. Consequently, the liner also heats the convective air flow. Conduction does not play a substantial role in heat removal from the canister surface. This natural circulation of air inside the vertical concrete cask (storage cask), in conjunction with radiation from the canister surface, maintains the fuel cladding temperature and all of the storage cask component temperatures below their design limits.

The thermal evaluation considers normal, off-normal, and accident conditions of storage. Each of these conditions can be described in terms of the environmental temperature, use of solar insolance, and the condition of the air inlet and outlet vents, as shown in Table 4.1-1. The design conditions for transfer are defined in Table 4.1-2.

This evaluation applies different component temperature limits and different material stress limits for long-term (steady-state) conditions and for short-term (transient) conditions. Normal storage is considered to be a steady-state condition. Off-normal and accident events, as well as the vacuum condition that temporarily occurs during the preparation of the canister while it is in the transfer cask and the time the canister, is in the transfer cask while filled with helium, are evaluated as transient conditions. The maximum allowable material temperatures for long-term and for transient conditions are provided in Table 4.1-3. The maximum component temperatures are provided in Table 4.1-4.

The NAC-MPC system is designed to store Yankee class spent fuel with a maximum heat load of 12.5 kW and reconfigured fuel assemblies with a maximum heat load of 0.102 kW per assembly. The temperature effects on the NAC-MPC storage cask and the canister, due to the reconfigured fuel assemblies, are bounded by the temperatures produced in the cask by the design basis fuel. Table 4.1-4 summarizes the results of the thermal evaluation. As shown in this table, the calculated temperatures are well below the allowable component temperatures for normal (long-term) storage conditions and for short-term events.

Table 4.1-1 Summary of Thermal Design Conditions for Storage

CONDITION	ENVIRONMENTAL TEMPERATURE (°F)	SOLAR INSOLANCE (1)	CONDITION OF STORAGE CASK VENTS
Normal	75	Yes	All vents open
Off-Normal - Half Air Inlets Blocked	75	Yes	Two inlets blocked
Off-Normal - Severe Heat	100	Yes	All vents open
Off-Normal - Severe Cold	-40	No	All vents open
Accident - Extreme Heat	125	Yes	All vents open
Accident - All Air Inlets and Outlets Blocked (2)	75	Yes	All vents blocked
Accident - Cask Burial Under Debris (3)	75	No	All vents blocked

(1) Solar Insolance per 10CFR71:
Curved Surface: 400 g cal/cm^2 (1475 Btu/ft^2) for a 12-hour period.
Flat Horizontal Surface: 800 g cal/cm^2 (2950 Btu/ft^2) for a 12-hour period.

(2) This condition bounds the case in which all inlets are blocked, with all outlets open.

(3) In the burial under debris condition, the inlets/outlets are blocked and, in addition, the debris is considered not to permit any heat transfer from the surface of the concrete. This is a highly conservative assumption.

Table 4.1-2 Summary of Thermal Design Conditions for Transfer

CONDITION(1)	**DURATION (Hours)**
Water Filled	20
Vacuum Drying(2)	10
Canister filled with Helium	36

(1) The canister is inside the Transfer Cask, with an ambient temperature of 75°F.

(2) The canister is filled with water for a maximum of 20 hours before the start of the vacuum drying process. The initial water temperature is considered to be 100°F.

Table 4.1-3 Maximum Allowable Temperature Limits (°F)

MATERIAL[1]	LONG-TERM	SHORT-TERM	REFERENCE
Concrete	150(B)/200(L)[2]	350	ACI 349
Zircaloy Fuel Cladding	644[3]	806[4]	PNL-6189 PNL-4835
Stainless Steel Fuel Cladding	644[3]	806	EPRI TR-106440
Aluminum Disk	650	700	MIL-HDBK-5G
NS-4-FR	300	300	Genden
Lead	600	600	Baumeister
SA693 Type 630 Stainless Steel	650	800	ASME B & PV Armco
SA240 Type 304 Stainless Steel	800	800	ASME B & PV
SA240 Type 304L Stainless Steel	800	800	ASME B & PV
ASTM A588 Carbon Steel	700	700	ASME Code Case N-71-17
ASTM A36 Carbon Steel	700	700	ASME Code Case N-71-17
BORAL Composite Sheet	850	1,000	AAR Advanced Structures

1. The minimum allowable temperature limit for all materials is -40°F.
2. B and L refer to bulk temperatures and local temperatures, respectively. The local temperature allowable applies to a restricted region where the bulk temperature allowable may be exceeded.
3. The temperature limits for 5-year and 10-year-cooled fuel are 380°C and 340°C, respectively. The lower value (340°C) is used.
4. The temperature limit for stainless steel-cladding is conservatively used for Zircaloy cladding.

Table 4.1-4 Summary of Thermal Evaluation for NAC-MPC Storage System

Design Conditions	Material Temperature (°F)									
	Concrete[1]	Fuel Clad	6061-T6 Al Alloy[2]	NS-4-FR[3]	Lead[3]	SA693 630 SS[2]	SA240 304 SS[4]	SA240 304L SS[4]	A36 Steel[1]	A588/A350LF2 Steel[3]
Allowable		Zr/SS								
Long-Term	150(Bulk) 200(Local)	644/644	650	300	600	650	800	800	700	700
Short-Term	350	806/806	700	300	600	800	800	800	700	700
Long-Term Conditions										
Normal (75°F Ambient)	133(Bulk) 165(Local)	563	527	N/A	N/A	529	183	319	165	N/A
Short-Term Conditions										
Off-Normal -Half Inlets Blocked (75°F Ambient)	168	565	529	N/A	N/A	531	192	318[5]	169	N/A
Off-Normal -Severe Heat (100°F Ambient)	196	587	552	N/A	N/A	554	213	347	196	N/A
Off-Normal -Severe Cold (-40°F Ambient)	5	453	411	N/A	N/A	412	44	187	5	N/A
Accident -Extreme Heat (125°F Ambient)	228	607	574	N/A	N/A	575	241	372	229	N/A
Transfer -Vacuum Drying	N/A	424	339	98	98	339	100	274	N/A	103
Transfer -Backfilled with Helium	N/A	597	569	188	191	570	140	430	N/A	237

1. Concrete cask components: Concrete cask and steel liner (A36).
2. Fuel basket components: Heat transfer disks (6061-T6) and support disks (SA 693, Type 630 stainless steel).
3. Transfer cask components: Shells (A588); bottom doors (A350LF2), neutron shield (NS-4-FR); and, gamma shield (lead).
4. Canister components: Shield lid (SA 240, Type 304 stainless steel) and shell, bottom plate and structural lid (SA 240 Type 304L stainless steel).
5. Although the maximum canister temperature is 1 degree lower than that of the normal condition, the overall canister temperature for the half inlet blocked condition is higher than that of the normal condition, which results in higher temperatures inside of the canister (fuel clad and disks).

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4.2 Summary of Thermal Properties of Materials

The thermal properties used in the thermal analyses are shown in Tables 4.2-1 through 4.2-11. The derivation of the effective conductivities is described in Sections 4.4.1.3 and 4.4.1.4.

Table 4.2-1 Thermal Properties of Solid Neutron Shield (NS-4-FR)

Property[1] (units)	**Value**
Conductivity (Btu/hr-in-°F)	0.0311
Density (lbm/in^3) (borated)	0.0589
Density (lbm/in^3) (nonborated)	0.0607
Specific Heat (Btu/lbm-°F)	0.39

1. Data developed by BISCO Products (NS-4-FR is now supplied by Genden Engineering Services and Construction Company. Genden has established a maximum continuous service temperature of 300°F.).

Table 4.2-2 Thermal Properties of Stainless Steels

Type 304 and Type 304L				
Property (units)	**Temperature (°F)[5]**			
	212	**392**	**572**	**752**
Conductivity[1] (Btu/hr-in-°F)	0.7800	0.8592	0.9333	1.0042
Density[1] (lbm/in^3)	0.2888	0.2872	0.2855	0.2839
Specific Heat[1] (Btu/lbm-°F)	0.1207	0.1272	0.1320	0.135
Emissivity[2]	0.36 at 300°F			
17-4PH, Type 630				
Property (units)	**Temperature (°F)[6]**			
	100	**200**	**500**	**700**
Conductivity[3] (Btu/hr-in-°F)	0.8417	0.8833	1.0167	1.1000
Density[4] (lbm/in^3)	0.284	0.284	0.284	0.284
Specific Heat[4] (Btu/lbm-°F)	0.11	0.11	0.11	0.11
Emissivity[2]	0.58	0.58	0.58	0.58

1. Hanford.
2. Bucholz.
3. "ASME Code Section II, Part D."
4. ARMCO.
5. Maximum Service Temperature: 800°F.
6. Maximum Service Temperature: 650°F long-term and 800°F short-term.

Table 4.2-3 Thermal Properties of Chemical Lead

Property (units)	Temperature (°F)[3]			
	209	400	581	630
Conductivity[1] (Btu/hr-in-°F)	1.6308	1.5260	1.2095	1.0079
Density[1] (lbm/in^3)	0.411	0.411	0.411	0.411
Specific Heat[1] (Btu/lbm-°F)	0.03	0.03	0.03	0.03
Emissivity[2]	0.28 at 75°F			

1 Edwards.

2 Baumeister.

3 Maximum service temperature: 600°F (based on preventing the lead from reaching its melting point of 620°F).

Table 4.2-4 Thermal Properties of Type 6061-T6 Aluminum Alloy

	Temperature (°F)^3					
Property (units)	**200**	**300**	**400**	**500**	**600**	**700**
Conductivity^3 (Btu/hr-in-°F)	8.25	8.38	8.49	8.49	8.49	8.49
Emissivity^2	0.22	0.22	0.22	0.22	0.22	0.22

1. ASME Code, Section II, Part D:
   The maximum temperature tabulated is 400°F. Since the conductivity increases as the temperature increases, using the value of 8.49 for higher temperatures is conservative (MIL-HDBK-5G, Figure 3.6.2.0).

2. Recommended value for aluminum in SCOPE (Bucholz) Version 1.2 Abbreviated Input Data Guide.

3. Maximum Service Temperature: 700°F, short-term service
   650°F, long-term service.

Table 4.2-5 Thermal Properties of Helium

Property (units)	Temperature (°F)			
	200	**400**	**600**	**800**
Conductivity[1] (Btu/hr-in-°F)	0.00808	0.00942	0.01075	0.01150
Density[1] (lbm/in^3)	4.83E-6	3.70E-6	3.01E-6	2.52E-6
Specific Heat[1] (Btu/lbm-°F)	1.24	1.24	1.24	1.24

[1] Kreith.

Table 4.2-6 Thermal Properties of Dry Air

Property (units)	**Temperature (°F)**			
	100	**300**	**500**	**700**
Conductivity[1] (Btu/hr-in-°F)	0.00128	0.00161	0.00193	0.0023
Density[1] (lbm/in^3)	4.11E-5	3.23E-5	2.38E-5	1.97E-5
Specific Heat[1] (Btu/lbm-°F)	0.240	0.244	0.247	0.253

[1] Kreith.

Table 4.2-7 Thermal Properties of Concrete

Property (units)	Temperature Range (°F)[5] 32 - 400
Conductivity[1] (Btu/hr-in-°F)	0.059
Specific Heat[1] (Btu/lbm-°F)	0.20
Density[2] (lbm/ft^3)	140
Emissivity[3,4]	0.90

(Emissivity = 0.93 for masonry, 0.94 for rough concrete; 0.9 is used)[3,4]

1 Fintel (Fig. 6-31, 0.71/12=0.059 Btu/hr. in.-°F).

2 ASTM-C150.

3 Kreith.

4 Siegel.

5 Maximum service temperature - long-term: 150°F bulk and 200°F local; short-term: 350°F. The local temperature allowable applies to a restricted region where the bulk temperature allowable may be exceeded.

Table 4.2-8 Thermal Properties of ASTM A 36, ASTM A 588 and ASTM A350 Carbon Steel

Property (units)	**Temperature (°F)**[5]				
	100	**200**	**400**	**500**	**700**
Conductivity[1] (Btu/hr-in-°F)	1.992	2.033	2.017	1.975	1.867
Density[2] (lbm/in^3)	0.284	0.284	0.284	0.284	0.284
Specific Heat[3] (Btu/lbm-°F)	0.113	0.113	0.113	0.113	0.113
Emissivity[4]	0.80	0.80	0.80	0.80	0.80

1 ASME Code, Section II, Part D, Table TCD.

2 Ross.

3 Kreith.

4 Baumeister.

5 Maximum service temperature: 700°F.

Table 4.2-9 Thermal Properties of Zircaloy and Zircaloy-4 Cladding

Property (units)	**Temperature (°F)**			
	392	**572**	**752**	**932**
Conductivity[1] (Btu/hr-in-°F)	0.69	0.73	0.80	0.87
Emissivity[1]	0.75	0.75	0.75	0.75

[1] NUREG/CR-0497 (minimum value of emissivity for a cladding surface).

Table 4.2-10 Thermal Properties of Fuel (UO_2)

Property (units)	Temperature (°F)			
	100	**440**	**570**	**793**
Conductivity[1] (Btu/hr-in-°F)	0.29	0.29	0.27	0.19

[1] NUREG/CR-0497 (The lower boundary of temperatures tabulated is 500°K [440°F]. Since the conductivity decreases as the temperature increases, using the value of 0.29 for the 100°F entry is conservative.).

Table 4.2-11 Thermal Properties of BORAL Composite Sheet

Property (units)	Temperature (°F)[3] 100	500
Conductivity[1] (Btu/hr-in-°F)		
Aluminum Clad[1]	7.805	8.976
Core Matrix[1]	4.136	3.698
Emissivity[1,2]	0.15	0.15

1 AAR Advanced Structures, standard specification for BORAL composite BRJREVO-940107.

2 The emissivity of the aluminum clad of the BORAL sheet ranges from 0.10 to 0.19 based on the BORAL specification. An averaged value of 0.15 is used.

3 Maximum service temperature: 850°F.

4.3 Specification of Components

There are three major components that must be maintained within their safe operating temperature ranges: the lead gamma shield and the NS-4-FR solid neutron shield in the transfer cask, and the aluminum heat transfer disk in the canister basket.

The safe operating ranges for the lead gamma shield, solid neutron shield and aluminum heat transfer disk are as follows:

Component	Safe Operating Range
Lead gamma shield	-40°F to +600°F
NS-4-FR solid neutron shield	-40°F to +300°F
Aluminum heat transfer disk	-40°F to +650°F (long-term); +700°F (short-term)

The safe operating range of the lead gamma shield is based on preventing the lead from reaching its melting point of 620°F (Baumeister).

The maximum operating temperature limit of the NS-4-FR solid neutron shield material to ensure sufficient neutron shielding capability is specified by the product supplier to be 300°F.

The safe operating range of the aluminum heat transfer disk is based on the integrity of the aluminum being maintained. The aluminum heat transfer disk is not a structural component to transfer load within the basket. Based on the MIL-HDBK-5F, aluminum at 700°F retains component performance. The maximum long-term and short-term operating temperatures for the aluminum heat transfer disk are taken to be 650°F and 700°F, respectively.

The maximum operating temperatures for other materials are shown in Table 4.1-3.

As shown in Tables 4.4.3-1 and 4.4.3-2, the maximum temperatures for these materials in the normal (long-term) conditions of storage are well below the allowable maximum temperatures. Maximum temperatures for off-normal and accident conditions are addressed in Chapter 11.

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4.4 Thermal Evaluation for Normal Conditions of Storage

4.4.1 Thermal Models

As listed below, six finite element models are utilized for the thermal evaluation of the NAC-MPC system for normal conditions of storage. The ANSYS program generates all models.

1. Two-Dimensional Axisymmetric Air Flow and Concrete Cask Model
2. Three-Dimensional Canister Model
3. Two-Dimensional Fuel Model
4. Two-Dimensional Fuel Tube Model
5. Three-Dimensional Transfer Cask and Canister Model
6. Two-Dimensional Reconfigured Fuel Assembly Model

The two-dimensional axisymmetric air flow and concrete cask model includes: the concrete storage cask, air in the air inlets, annulus and the air outlets, and the canister shell. It is used to perform computational fluid dynamic analyses to determine the mass flow rate, velocity and temperatures of the air flow, as well as the temperature distribution of the concrete, concrete cask steel liner and the canister shell.

The three-dimensional canister model comprises the fuel assemblies, fuel tubes, stainless steel support disks, aluminum heat transfer disks, the canister shell, lids and bottom plate. The canister model is employed to evaluate the temperature distribution of the fuel cladding and components of the canister and basket. The fuel regions and the fuel tubes with BORAL plates in the three-dimensional canister model are modeled using effective conductivities.

The effective conductivity of the fuel is determined using the two-dimensional fuel model, which is a detailed two-dimensional thermal model of the fuel assembly. The model includes the fuel pellets, cladding and gas (considered to be helium) occupying the space between the fuel rods and in the gap between the fuel pellets and the fuel rod cladding.

The two-dimensional fuel tube model is used to determine the effective conductivities of the tube wall and BORAL plate.

The three-dimensional transfer cask and canister model comprises the transfer cask, the canister and the canister internals—i.e., the three-dimensional canister model with the transfer cask added. This model is used to perform transient analysis for the transfer condition when the canister is in the vacuum condition during drying.

The two-dimensional reconfigured fuel assembly model comprises the fuel rods, fuel tubes, the shell casing (the square tube with the same external dimensions as an intact fuel assembly) and the gas (helium) occupying the gap between fuel rod and tube, the space between fuel tubes and the gap between shell casing and the fuel assembly tube. This model is used to determine the temperature distribution of the reconfigured fuel assembly.

These models are described in Section 4.4.1.1 through 4.4.1.6.

4.4.1.1 Two-Dimensional Axisymmetric Air Flow and Concrete Cask Model

The storage cask consists of the fuel canister, concrete, steel liner, air gap between the fuel canister shell and steel liner, and air inlets/outlets through the concrete region. The fuel canister with a design heat load of 12.5 kW will be cooled by (1) natural/free convection of air through the lower vents, the vertical air annulus, and the upper vents; and (2) radiant heat transfer between the surfaces of the canister shell and the steel liner. The heat transferred to the liner will be rejected by air convection in the annulus and by conduction through the concrete. The heat flow through the concrete will be dissipated to the surroundings by natural convection and radiant heat transfer. The temperature in the concrete region is controlled by radiant heat transfer between the vertical annulus surfaces (canister shell outer surface and steel liner inner surface), natural convection of air in the annulus, and boundary conditions applicable to the storage cask outer surfaces—e.g., natural convection and radiant heat transfer between the outer surfaces and the environment, including consideration of incident solar energy. These heat transfer modes are combined in the air flow and concrete cask model. The entire thermal system, including mass, momentum, and energy, is analyzed using ANSYS FLOTRAN.

4.4.1.1.1 Finite Element Model for Air Flow in Storage Cask

The storage cask has four air inlets at the bottom and four air outlets at the top that extend through the concrete. Since the configuration is symmetrical, it can be simplified into a two-dimensional axisymmetric model, as shown schematically in Figure 4.4.1.1-1, by using equivalent dimensions for the air inlets and outlets, which are assumed to extend around the storage cask periphery. This model consists of the following regions: canister shell, steel liner, concrete, air inlet, transitional region, vertical annulus, and air outlet. The canister shell is included in this model in order to apply the heat flux for the design fuel heat. The canister model is described in Section 4.4.1.2.

The two-dimensional axisymmetric air flow and concrete cask finite element model is shown in Figure 4.4.1.1-2. The model has the following features. In the air region, only quadrilateral elements are used and the element sizes are nonuniform with much smaller element sizes close to the walls. In other regions to simulate conduction, a mix of quadrilateral elements and triangular elements are used.

In this model, the radiation heat transfer across the vertical air annulus is also included.

4.4.1.1.2 Loads and Boundary Conditions

In the normal storage condition, the concrete cask has the following loads and boundary conditions:

1. Heat flux from the active fuel region.

Since only the canister shell is included in this air flow model, an equivalent nonuniform heat flux from the active fuel region is applied to the inner surface of the canister shell. This heat flux corresponds to 12.5kW, which is the heat generated by the spent fuel. The distribution of the heat flux is based on the axial power distribution, as described in Section 5.2.3, for the design-basis fuel (Figure 4.4.1.1-3).

2. Solar insolance to the outer surfaces of the storage cask.

The solar insolance to the storage cask outer surface is considered in the model. The incident solar energy is applied based on 24-hour averages as shown below.

Side surface: $$\frac{1475 \text{Btu}/\text{ft}^2}{24 \text{hrs}} = 61.46 \text{Btu}/\text{hr}\cdot\text{ft}^2$$

Top surface: $$\frac{2950 \text{Btu}/\text{ft}^2}{24 \text{hrs}} = 122.92 \text{Btu}/\text{hr}\cdot\text{ft}^2$$

3. Natural convection heat transfer at the outer surfaces of the storage cask.

Natural convection heat transfer at the outer surfaces of the storage cask is evaluated using the heat transfer correlation for vertical and horizontal plates (Kreith, Incropera). This method assumes a surface temperature and then estimates Grashof (Gr) or Rayleigh (Ra) numbers to determine whether correlation for a laminar model or for a turbulence model should be used. Since Grashof or Rayleigh numbers are much higher than the critical values, correlation for a turbulence model is used as shown in the following.

Side surface (Kreith):

$$\begin{aligned} Nu &= 0.13(Gr \cdot Pr)^{1/3} \\ h_c &= Nu \cdot k_f / H_{VCC} \end{aligned} \quad \text{for } Gr > 10^9$$

Top surface (Incropera):

$$\begin{aligned} Nu &= 0.15 Ra^{1/3} \\ h_c &= Nu \cdot k_f / L \end{aligned} \quad \text{for } Ra > 10^7$$

where:

h_c	Average natural convection heat transfer coefficient
H_{vcc}	Height of the storage cask
Gr	Grashof number
k_f	Conductivity
L	Top surface characteristic length, L = area / perimeter
Nu	Average Nusselt number
Pr	Prandtl number
Ra	Rayleigh number

All material properties required in the above equations are evaluated based on the film temperature—that is, the average value of the surface temperature and ambient temperature.

4. Radiation heat transfer at the storage cask outer surfaces.

The radiation heat transfer between the outer surfaces and ambient is evaluated in the model by calculating an equivalent radiation heat transfer coefficient.

$$h_{rad} = \frac{\sigma(T_1^2 + T_2^2)(T_1 + T_2)}{1/\varepsilon_1 + 1/\varepsilon_2 + 1/F_{12} - 2}$$

where:

h_{rad}	Equivalent radiation heat transfer coefficient
F_{12}	View factor
T_1 and T_2	Surface (T_1) and ambient (T_2) temperatures (°K)
ε_1 and ε_2	Surface (ε_1) and ambient (ε_2=1) emissivities
σ	Stefan-Boltzmann Constant

At the storage cask side surface, an emissivity for concrete surface of ε_1 = 0.9 is used and a calculated view factor (F_{12}) = 0.197 (Chapter 13 of Incropera) is applied. Since the center cask is surrounded by other casks, its view factor to the ambient is reduced.

At the top, an emissivity of ε_1 = 0.8 for a carbon steel surface is used, and a view factor (F_{12}) = 1 is applied.

5. Other flow and thermal boundary conditions.

At all walls bounding the air flow region, a zero velocity, vx=vy=0, is applied. At the inlet and outlet surfaces, a relative zero pressure, $p_{in} = p_{out} = 0$, is applied. The inlet temperature of the air is reasonably assumed to be the same as the ambient temperature. The effect of surrounding loaded casks on air inlet temperature is considered to be negligible because of the 15-foot center to center spacing of the casks (approximately 4.5 feet, clear opening). The natural air flow patterns between the casks negate potential stagnation effects.

4.4.1.1.3 Accuracy Check of the Numerical Simulation

To ensure the accuracy of the numerical simulation on the air flow in the storage cask and to ensure reliable numerical results, the following checks and confirmations have been performed.

1. Global convergence of the iteration process for the nonlinear system.

The system controlling air flow through the cask and, therefore, the temperature field is nonlinear and is solved iteratively. The global iteration process is monitored by checking the variation of parameters with the global iteration—e.g., the maximum air temperature, the mass flow rate, and the heat carried out of the storage cask by air convection. The mass flow rate varying with the global iteration is shown in Figure 4.4.1.1-4. As shown, after 10,000 global iterations, the mass flow rate is constant, indicating that the global iteration process has converged. At the converged state, the mass flow rate at the air inlets agrees with that at the air outlets. All of the results presented are at the converged state.

2. Finite element mesh adequacy study.

Element size or number of elements used in the simulation will affect the accuracy and reliability of the numerical prediction, even though converged results can be obtained.

Two tests, using different element sizes, have been performed for the normal case. Results are shown in Table 4.4.1.1-1, together with the number and size of elements close to the vertical annulus surfaces. As shown, results for the two test cases are in agreement, with the maximum difference being less than 3.3 percent in net heat carried out of the cask by air (Q_{air}).

3. Overall energy balance and mass balance.

This step validates the overall energy balance and mass balance. The mass balance is shown in Figure 4.4.1.1-4. At the converged state, the mass flow rate at the air inlets matches the mass flow rate at the air outlets, showing that an excellent mass balance has been obtained.

The overall energy balance is checked by computing the total heat input (Q_{in}) and total heat output (Q_{out}). The total heat input includes the total heat from the fuel (Q_{fuel}) and the total solar energy (Q_{sun}) incident on the storage cask outer surfaces. The total heat output is the sum of heat carried out of the cask by air (Q_{air}) and by convection and radiation heat loss at the storage cask outer surfaces (Q_{con}). During the normal storage condition:

$$Q_{in} = Q_{fuel} + Q_{sun} = 12.5\text{kW} + 9.85\text{kW} = 22.35\text{kW}$$

$$Q_{out} = Q_{air} + Q_{con} = 12.43\text{kW} + 10.25\text{kW} = 22.68\text{kW}$$

$$Q_{out}/Q_{in} = 1.015$$

Therefore, the overall energy balance is demonstrated to be within 1.5 percent.

Figure 4.4.1.1-1 Two-Dimensional Axisymmetric Air Flow and Concrete Cask Model

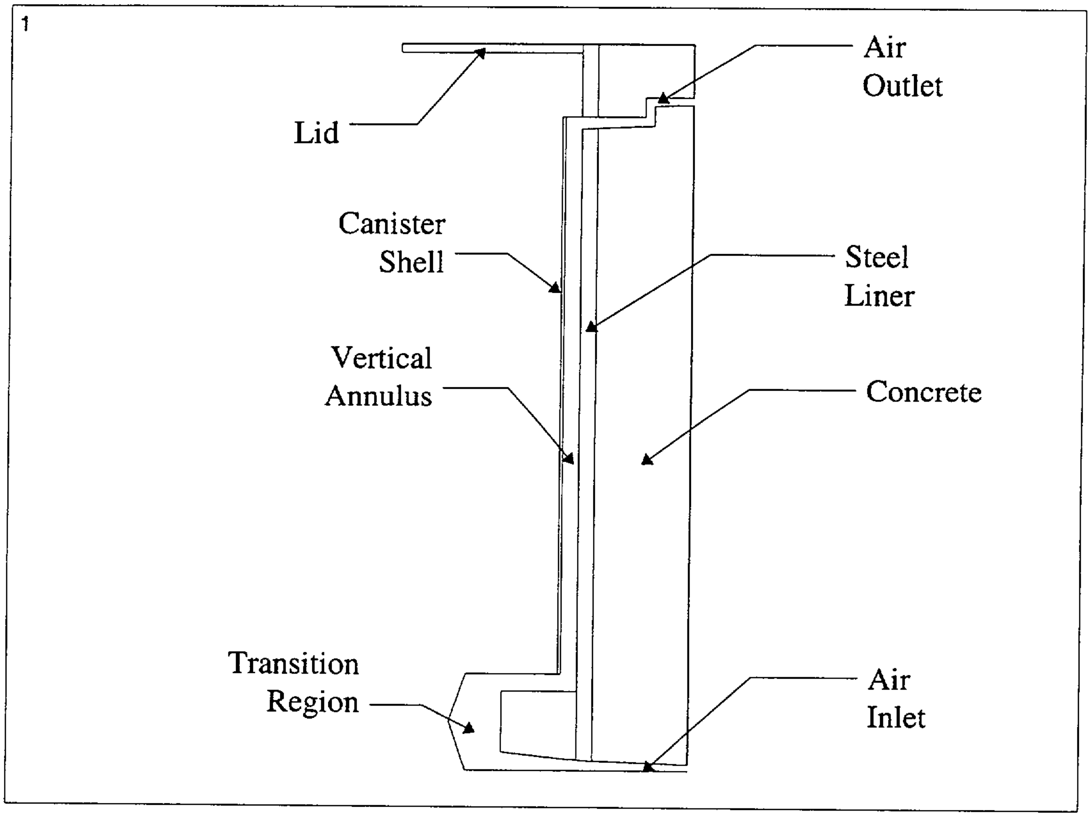


Figure 4.4.1.1-2 Two-Dimensional Axisymmetric Air Flow and Concrete Cask Finite Element Model

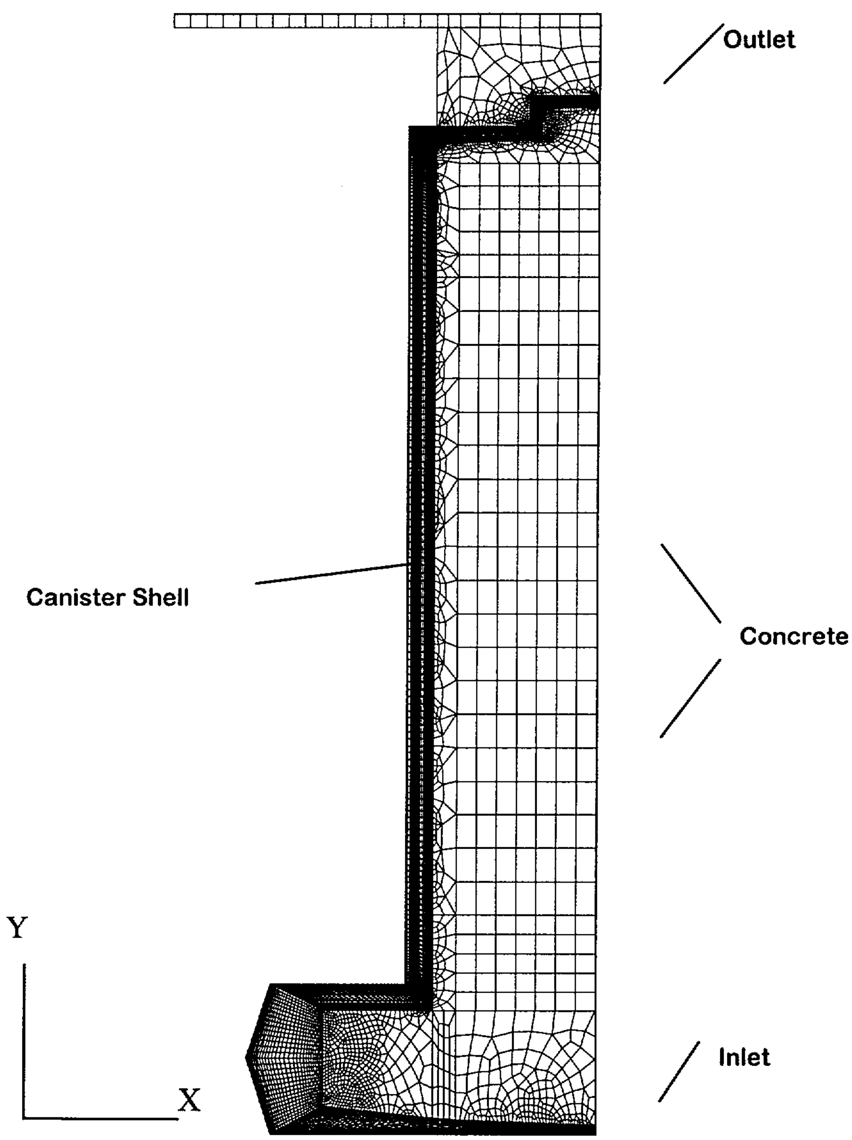


Figure 4.4.1.1-3 Axial Power Distribution for Yankee Class Fuel

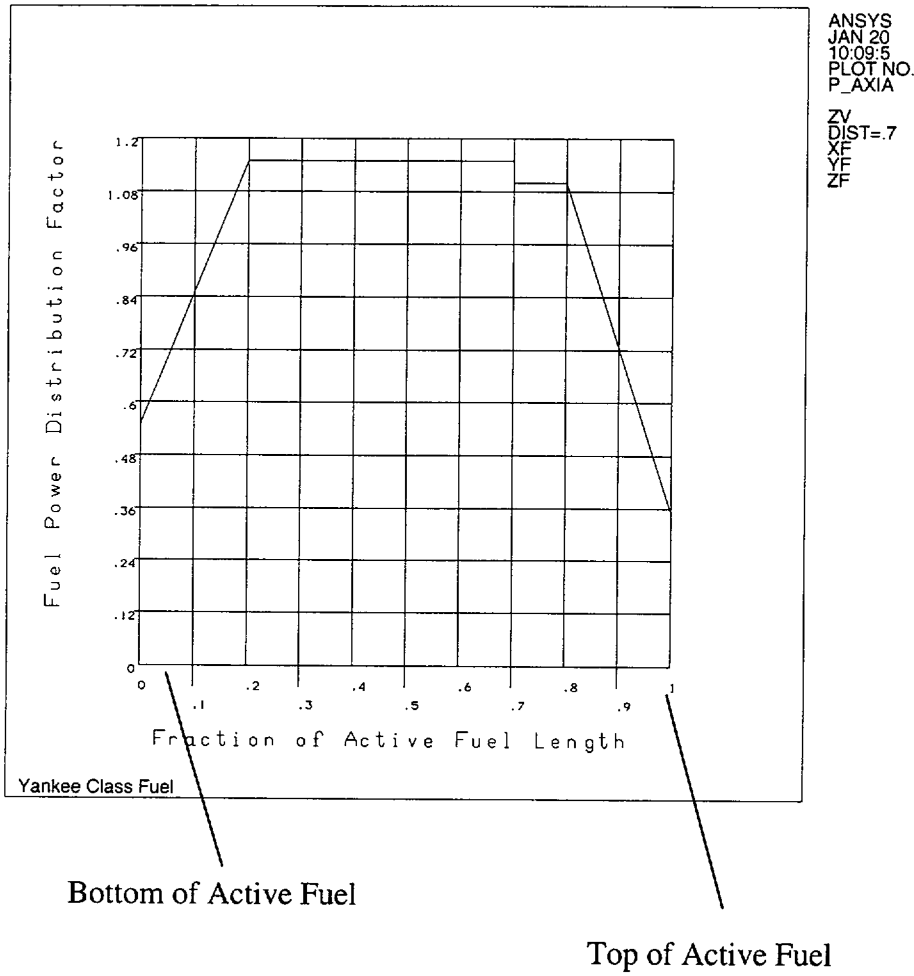


Figure 4.4.1.1-4 Convergence Process of Mass Flow Rate

Mass in: mass flow rate at the air inlets, kg/s

Mass out: mass flow rate at the air outlets, kg/s

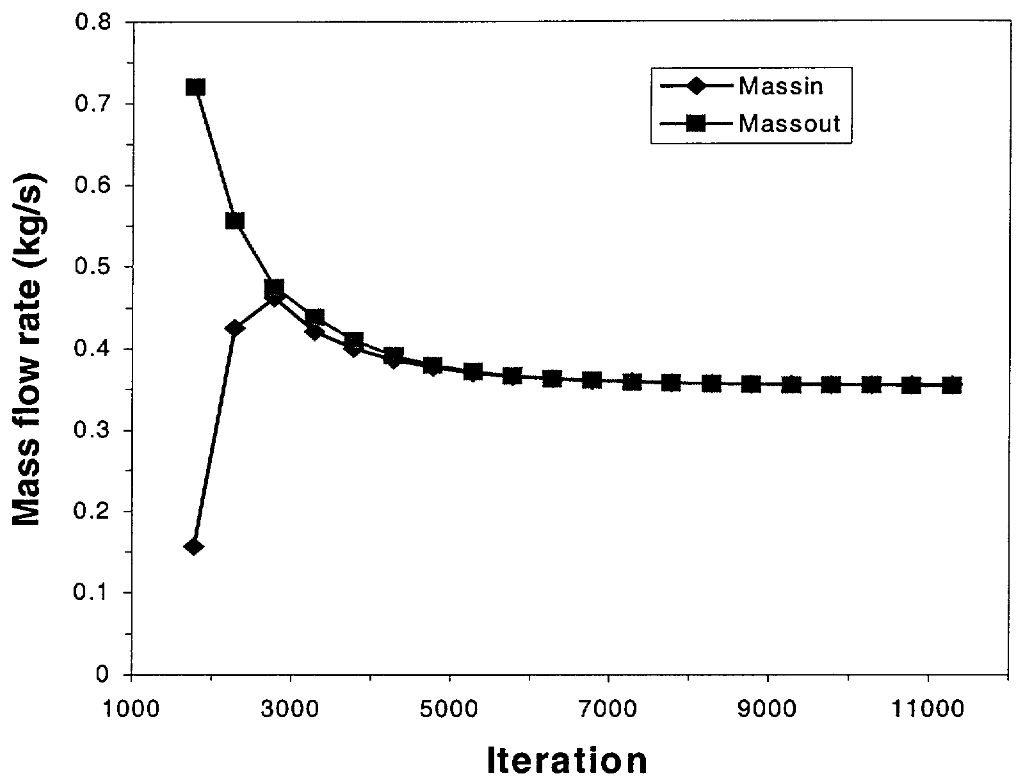


Table 4.4.1.1-1 Comparison of Numerical Results Using Different Element Sizes and Number of Elements

Test Case	Element Number	Element Size(m)	$Tmax_{air}$ (K)	$Tmax_{con}$ (K)	Mass (kg/s)	Vx_{out} (m/s)	T_{out} (K)	Q_{air} (kW)
ES1	12167	0.00184	433.25	345.33	.3582	.7780	332.84	12.84
ES2	15631	0.00136	430.61	346.89	.3543	.7673	332.45	12.43
ES1/ES2	.7784	1.353	1.0061	.9955	1.011	1.014	1.0012	1.033

where:

$Tmax_{air}$	Maximum air temperature
$Tmax_{con}$	Maximum concrete temperature
Element size	Refer to the element adjacent to the wall of the vertical gap
Mass	Average mass flow rate of air between air inlet and outlet
Vx_{out}	Average velocity component in x-direction
T_{out}	Average air temperature at the air outlet
Q_{air}	Net heat carried out of the cask by air

4.4.1.2 Three-Dimensional Canister Model

The three-dimensional canister model is shown in Figures 4.4.1.2-1 and 4.4.1.2-2. ANSYS SOLID70 three-dimensional conduction elements and LINK31 radiation elements are used to construct the model. The model includes: the fuel assemblies, fuel tubes, support disks, heat transfer disks, the canister shell, lids, bottom plate and helium. Based on symmetry, only half of the canister is modeled. As shown in Figure 4.4.1.2-1, the interior of the canister contains the active fuel region.. No conduction elements are defined outside of this region in the axial direction; i.e., top and bottom fittings of the fuel assemblies, fuel tubes enclosing the top and bottom fittings, the first support disk (counted from the top end), and the top and bottom weldments are not included in the model. Conduction through these components is conservatively ignored. The canister shell temperatures obtained from the two-dimensional axisymmetric air flow and concrete cask model (Section 4.4.1.1) are applied at the canister shell surface in the model as boundary conditions. The top surface of the canister lid and the bottom surface of the canister bottom plate in the model are considered to be adiabatic. In the model, the fuel assemblies are considered to be centered in the fuel tubes. The fuel tubes are centered in the slots of the support disks and heat transfer disks. The basket is centered in the canister. These assumptions are conservative, since any contact between components will provide a more efficient path to reject the heat.

This model includes the following gaps:

LOCATION	GAP SIZE
1. Gap between the support disk and the canister shell	0.205 inch
2. Gap between the heat transfer disk and the canister shell	0.345 inch
3. Gap between the exterior surface of the fuel tube and the inside surface of the slots in the disks	0.079 inch
4. Gap between the BORAL sheet and tube/cladding	0.003 inch

Gas inside the canister is modeled as helium. Note that the nominal gap (room temperature) size between the support disk and the canister shell is 0.12 inch, and the nominal gap size between the heat transfer disk and the canister shell is 0.26 inch. As shown above, the thermal model conservatively considers these gaps to be 0.205 inch and 0.345 inch (at room temperature) for the support disk and heat transfer disk, respectively. Considering the differential thermal expansion of the disks and the canister shell, these gap sizes are adjusted for the solution of the thermal analysis. In the model, the gap between the support disk and the canister shell is increased to 0.22 inch, and the gap between the heat transfer disk and the canister shell is reduced to 0.295 inch. These gaps are conservative and remain bounding even if the worst case fabrication tolerances are considered. The structural lid and the shield lid are expected to be in full contact due to the weight of the structural lid. The thermal resistance across the contact surface is considered to be negligible and, therefore, no gap is modeled between the lids.

All material properties used in the model, except the effective properties discussed below, are shown in Tables 4.2-1 through 4.2-11.

The fuel regions (inside tubes) are modeled as homogenous regions with effective conductivities, determined by the two-dimensional fuel model as described in Section 4.4.1.3. The center slot of the basket contains no fuel and is modeled as helium. The fuel tube and the BORAL plate, including helium gaps on both sides of the BORAL sheet and the gap between the stainless steel cladding and the disks, are modeled as one-element thick with effective conductivities, established using the two-dimensional tube model described in Section 4.4.1.4.

In this model, radiation heat transfer is taken into account in the following locations:

1. From the top of the fuel region to the bottom surface of the shield lid.
2. From the bottom of the fuel region to the top surface of the canister bottom plate.
3. From the exterior surfaces of the fuel tubes (between SS and AL disks) to the inner surface of the canister shell.
4. Across all four gaps, described above.

Radiation elements (LINK31) are used to model the radiation effect for the first three locations. Radiation across gaps is accounted for by establishing effective conductivities for the gas in the gap, as shown below. Since the gaps represented in the model are small compared to the surfaces separated by the gaps, the view factor is taken to be unity.

Radiation heat transfer between two nodes i (hotter node) and j (colder node) is accounted for by the expression:

$$q_r = \sigma \varepsilon_{eff} AF\left(T_i^4 - T_j^4\right)$$

where:

σ = the Stefan-Boltzman constant
= 1.19×10^{-11} Btu/hr-in^2-°R^4
ε_{eff} = effective gray body emissivity
A = surface area
F = the gray body shape factor for the surfaces
T_i = temperature of the i th node
T_j = temperature of the j th node

The total heat transfer can be expressed as the sum of the radiation and the conduction processes.

$$Q_t = q_r + q_k$$

where q_r is specified above for the radiation heat transfer and q_k, which is the heat transfer by conduction is expressed as:

$$q_k = \frac{KA}{g} 1(T_i - T_j)$$

where:

T_i = temperature of the i th node
T_j = temperature of the j th node
g = gap distance (between the two surfaces defined by node i and node j)
K = conductivity of the gas in the gap
A = area of gap surface

By combining the two expressions (for q_k and q_r) and factoring out the term $A(T_i - T_j)/g$,

$$Q_t = [g\sigma\varepsilon_{eff}F(T_i^2 + T_j^2)(T_i+T_j) + K][A(T_i - T_j)/g]$$

or

$$Q_t = K_{eff}A(T_i - T_j)/g$$

where:

$$K_{eff} = g\sigma\varepsilon_{eff}F(T_i^2 + T_j^2)(T_i+T_j) + K$$

The material conductivity used in the analysis for the elements comprising the gap includes the heat transfer by both conduction and radiation.

Effective emissivities are used for all radiation calculations, based on the formula below (Kreith).

$\varepsilon_{eff} = 1/\,(1/\varepsilon_1 + 1/\varepsilon_2 - 1)$ where ε_1 and ε_2 are the emissivities of two parallel plates

Radiation between the exterior surfaces of the fuel tubes and the radiation between the stainless steel support disk and the aluminum disk are conservatively ignored in this model.

Volumetric heat generation ($Btu/hr\text{-}in^3$) is applied to the active fuel region based on a total heat load of 12.5 kW, active fuel length of 91 inches and an axial power distribution as shown in Figure 4.4.1.1-3.

Figure 4.4.1.2-1 Three-Dimensional Canister Model

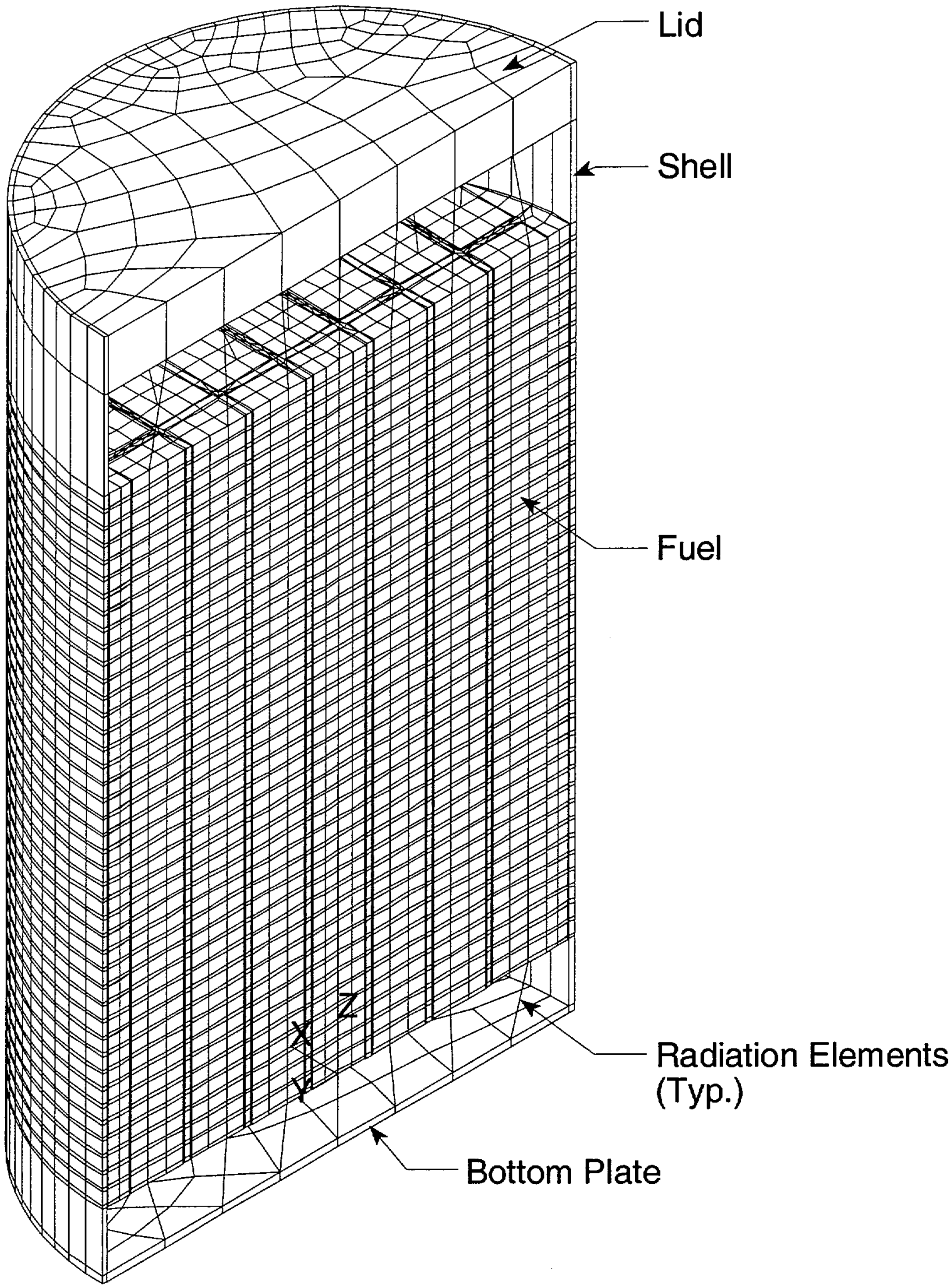
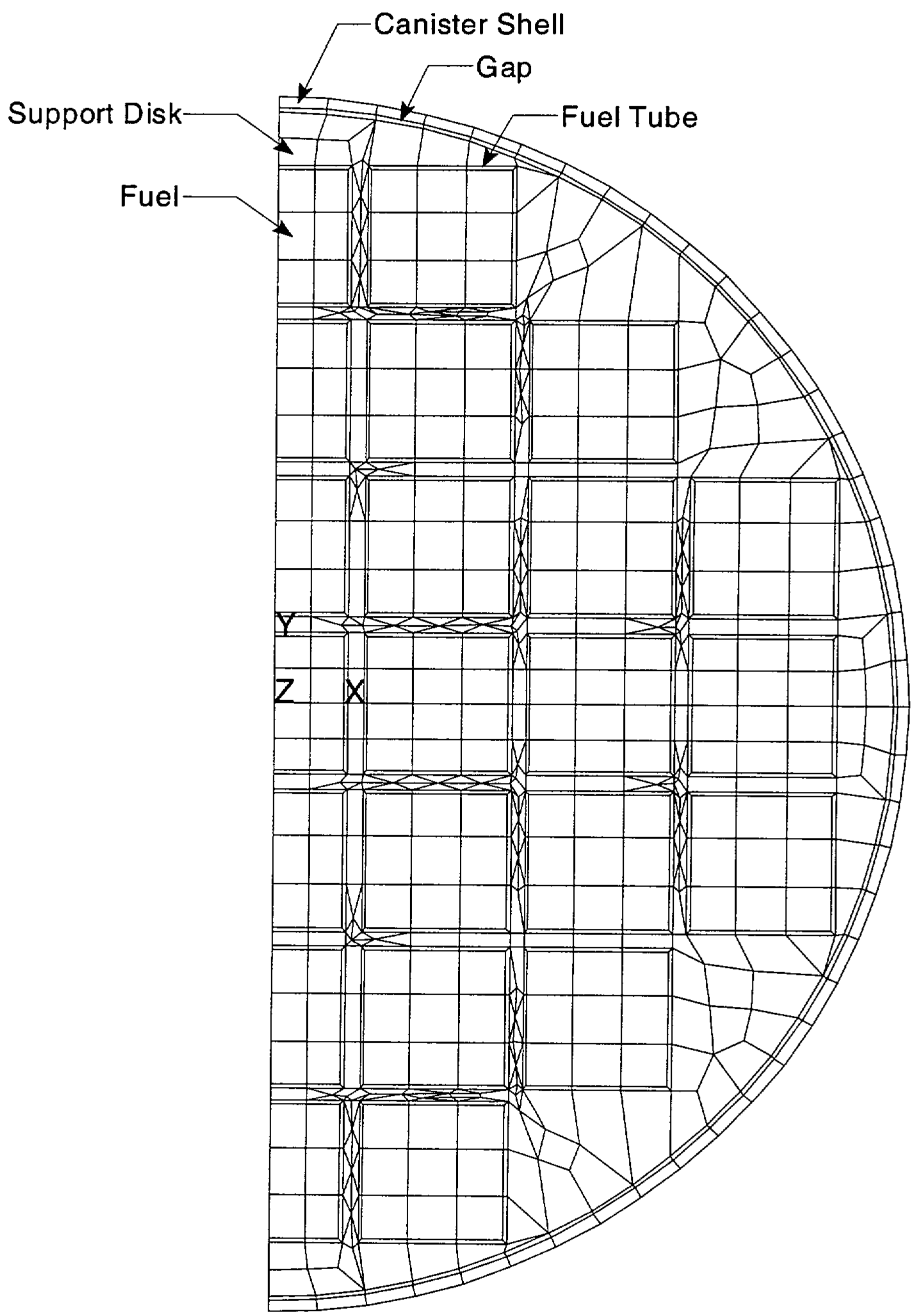


Figure 4.4.1.2-2 Three-Dimensional Canister Model - Cross-Section



4.4.1.3 Two-Dimensional Fuel Model

The effective conductivity of the fuel is determined by a two-dimensional finite element model of the fuel assembly. The model includes: the fuel pellets, cladding, gas between fuel rods and gas (considered to be helium) occupying the gap between the fuel pellets and cladding. The configuration of the design basis fuel (CE Type A) is used in the model. Note that the fuel cladding material for this fuel is zircaloy, which is less conductive than stainless steel. Therefore, the fuel model bounds the fuel with zircaloy cladding, as well as the fuel with stainless steel cladding. Modes of heat transfer modeled include conduction and radiation between individual fuel rods for the steady-state condition. The model is shown in Figure 4.4.1.3-1.

ANSYS PLANE55 conduction elements and LINK31 radiation elements are used in the model, which includes a total of 240 fuel rods. Each fuel rod consists of the pellet, zircaloy cladding, and a gap between the pellet and cladding. The gas in the gap between the pellet and cladding, as well as the gas between fuel rods, is considered to be helium. Radiation elements are defined between fuel rods and from the fuel rods to the boundary of the model (inside surface of the fuel tube). Radiation effects at the gap between the pellets and the cladding are conservatively ignored. Effective emissivities are determined using the formula shown in Section 4.4.1.2.

The effective conductivity for the fuel is determined using the equation (SANDIA) to determine the maximum temperature of a square cross-section of an isotropic homogeneous fuel with a uniform volumetric heat generation. At the boundary of the square cross-section, the temperature is constrained to be uniform. The expression for the maximum temperature is given by:

$$T_c = T_e + 0.29468 \, (Q \, a^2 / K_{eff})$$

where:

T_c = the temperature at the center of the fuel (°F)
T_e = the temperature applied to the exterior of the fuel (°F)
Q = volumetric heat generation rate (Btu/hr-in^3)
a = half length of the square cross-section of the fuel (inch)
K_{eff} = effective thermal conductivity for the isotropic homogeneous fuel material (Btu/hr-in-°F)

Volumetric heat generation (Btu/hr-in^3) based on the design heat load of 12.5 kW is applied to the pellets. The effective conductivity is determined based on the heat generated and the temperature difference from the center of the model to the edge of the model. The temperature-dependent effective properties, as shown below, are established using different boundary temperatures. The effective conductivity in the axial direction of the fuel assembly is calculated based on the material area ratio.

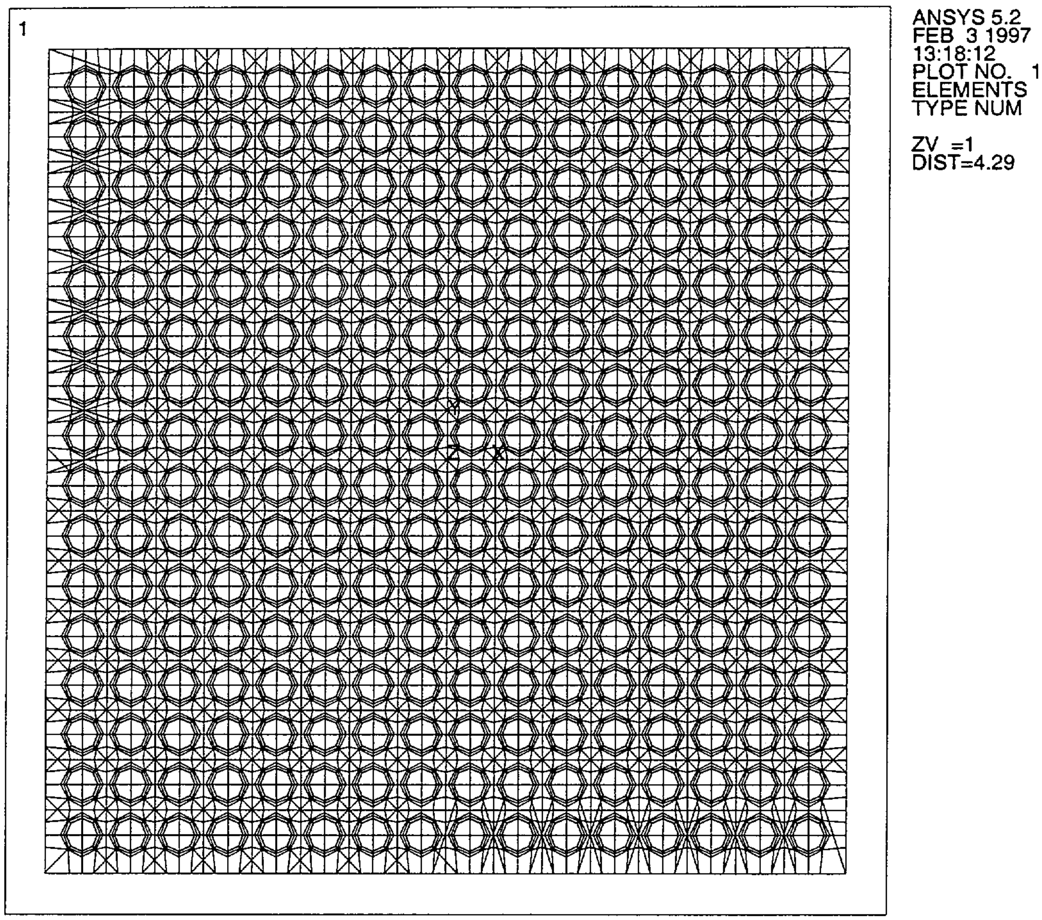
Conductivity (Btu/hr-in-°F)

Temperature (°F)	kxx	kyy	kzz
128	0.0167	0.0167	0.163
322	0.0204	0.0204	0.152
518	0.0262	0.0262	0.141
714	0.0330	0.0330	0.138
911	0.0405	0.0405	0.140

Note:

1. x and y axes are in-plane of the model, z is in the canister axial direction.
2. The temperature associated with each row is the average temperature of the fuel assembly.

Figure 4.4.1.3-1 Two-Dimensional Fuel Model



4.4.1.4 Two-Dimensional Fuel Tube Model

The purpose of the two-dimensional fuel tube model is to determine the effective conductivity of the fuel tube and BORAL plate, which is used in the three-dimensional canister model. As shown in Figure 4.4.1.4-1, this model includes the fuel tube, the BORAL plate (including the core matrix sandwiched by aluminum claddings), helium gaps on both sides of the BORAL plate, and the helium gap between the stainless steel cladding on the outside of the BORAL plate and the support disk or heat transfer disk.

ANSYS PLANE55 conduction elements and LINK31 radiation elements are used to construct the model. The model consists of eight layers of conduction elements and six radiation elements that are defined at the helium gaps (two for each gap). The thickness of the model (x-direction) is the distance measured from the inside face of the fuel tube to the inside face of the slot in the support disk (assuming the fuel tube is centered in the hole in the disk). The tolerance of the BORAL plate thickness, 0.003 inch, is used as the gap size for both sides of the BORAL plate. The height of the model is defined as equal to the width of the model.

Heat flux is applied at the left side of the model (fuel tube), and the temperature at the right boundary of the model is constrained. The heat flux is determined based on the design heat load of 12.5 kW. The maximum temperature of the model (at the left boundary) and the temperature difference (ΔT) across the model are calculated by ANSYS. The effective conductivity is determined using the following formula:

$$q = k\,(A/L)\,\Delta T$$

or

$$k = q\,L\,/\,(A\,\Delta T)$$

where:

q = heat rate (Btu/hr)
A = area (in^2)
L = length of model (in)
ΔT = temperature difference across the model (°F)
k = effective conductivity (Btu/hr-in-°F)

The temperature-dependent conductivity (k) is determined by varying the temperature constraints at one boundary of the model and resolving for the heat rate (q) and temperature difference. The effective conductivity for the parallel path (the Y direction in Figure 4.4.1.4-1) is calculated by:

$$\text{Kyy} = \frac{\sum K_i t_i}{T}$$

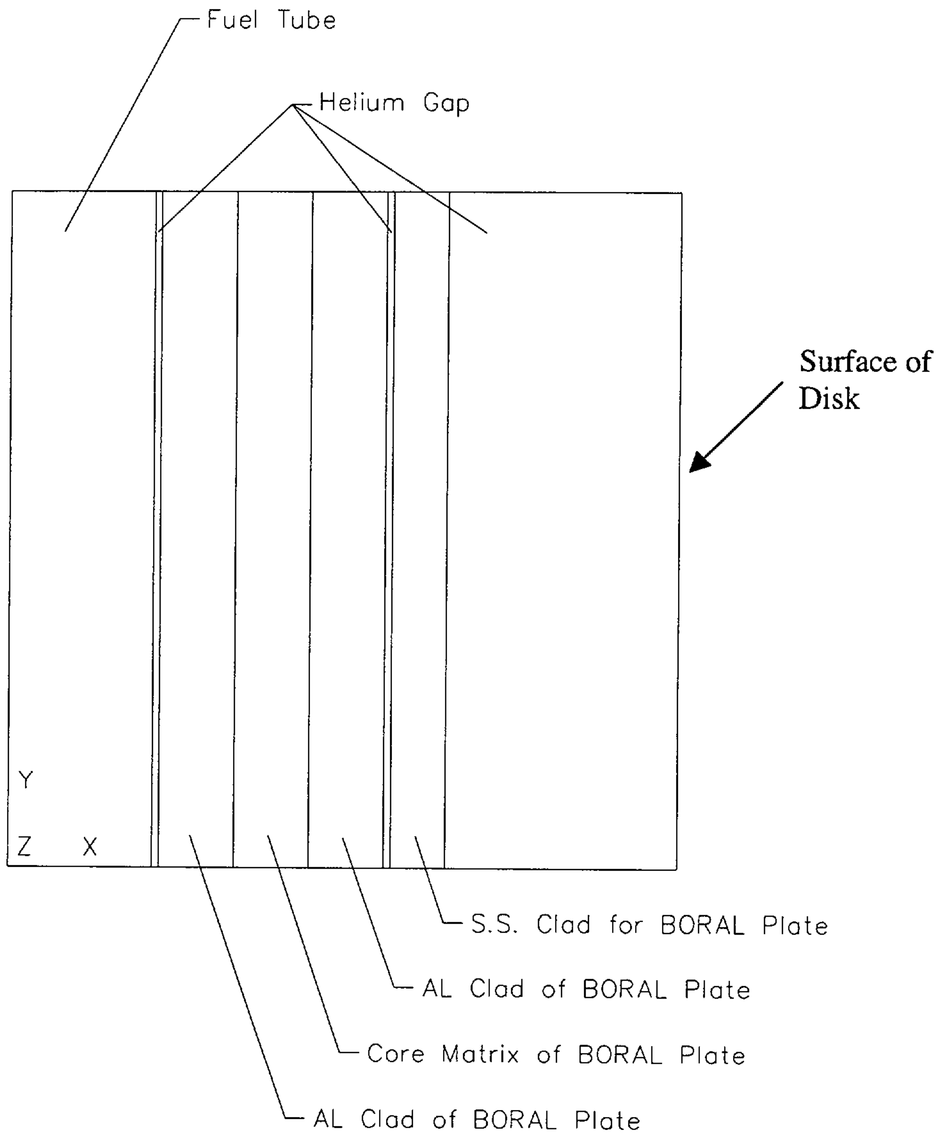
where:

K_i = thermal conductivity of each layer (fuel tube, helium aluminum clad)

t_i = thickness of each layer

T = total thickness of fuel tube, gaps (in Figure 4.4.1.4-1)

Figure 4.4.1.4-1 Two-Dimensional Fuel Tube Model



4.4.1.5 Three-Dimensional Transfer Cask and Canister Model

The three-dimensional transfer cask and canister model comprises the transfer cask, the canister and the canister internals. This model is one half (90°) of the three-dimensional canister model (Section 4.4.1.2) with the transfer cask added. The region below the active fuel is modeled with solid elements. The radiation from the bottom of the fuel to the top of the canister bottom plate is considered using the method described in Section 4.4.1.2. The trunnions and the retaining ring at the top of the transfer cask are not included in the model. This is conservative, since it reduces the surface area for radiation and convection. The model is used to calculate the transient temperature distribution for the fuel cladding and the components of the transfer cask, canister and basket for the transfer condition when the canister is back-filled with helium. The model is shown in Figure 4.4.1.5-1. ANSYS SOLID70 elements and LINK31 elements are used to construct the model. Convection and radiation heat transfer are considered at the surfaces of the transfer cask and on the top of the canister lid. The bottom of the transfer cask is conservatively considered to be adiabatic. An ambient temperature of 75°F is assumed and no solar insolance is considered, since the transfer operation occurs inside a building. The canister is considered to be centered in the cavity of the transfer cask. In addition to the gaps inside the canister as described in Section 4.4.1.2, the following two gaps are considered in the model:

	LOCATION	GAP SIZE
1.	Gap between transfer cask inner shell and lead	0.063 inch
2.	Gap between the canister outer surface and the transfer cask inner shell	0.43 inch

These two gaps are considered to be filled with air. The 0.063-inch gap size between the transfer cask inner shell and the lead is used based on the dimensional tolerances of the transfer cask design. The gap size of 0.43 inch is based on the nominal dimensions of canister shell and the transfer cask. Radiation heat transfer across the gaps is considered using the same method described in Section 4.4.1.2. Radiation at the transfer cask outer surface and canister lid top surface is accounted for by the expression:

$$q_r = \sigma \varepsilon_{eff} AF\left(T_i^4 - T_j^4\right)$$

where:

σ = the Stefan-Boltzman constant
= 1.19×10^{-11} Btu/hr-in^2-°R^4

ε_{eff} = emissivity

A = surface area

F = the gray body shape factor for the surfaces

T_i = temperature of the hotter node

T_j = temperature of the colder node

Radiation heat transfer from the surface can be incorporated in the model by modifying the convection coefficient as shown below:

$$Q_t = q_r + q_c$$

where q_r is specified above for the radiation heat transfer and q_c, which is the heat transfer by convection, is expressed as:

$$q_c = h_c A(T_i - T_j)$$

where:

h_c = film coefficient (Btu/hr-in²)

The q_r can be rewritten as:

$$q_r = \sigma \varepsilon_{eff} AF(T_i^2 + T_j^2)(T_i + T_j)(T_i - T_j)$$

By combining both expressions:

$$Q_t = (\sigma \varepsilon_{eff} F(T_i^2 + T_j^2)(T_i+T_j) + h_c)A(T_i - T_j)$$

or

$$Q_t = h_{eff} A(T_i - T_j)$$

where:

$$h_{eff} = \sigma \varepsilon_{eff} F(T_i^2 + T_j^2)(T_i + T_j) + h_c$$

The convection coefficient, h_{eff}, used for the cask surface now includes the radiation heat transfer. The form factor (F) is taken to be unity.

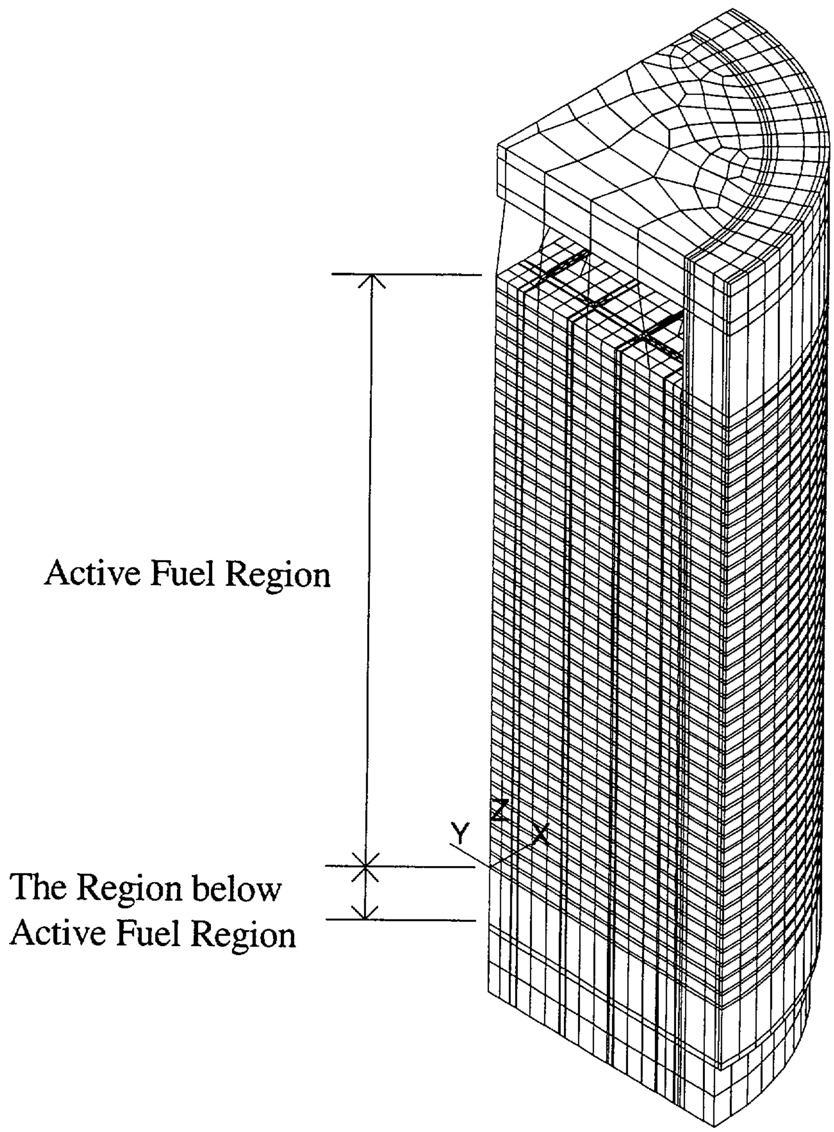
The convection heat transfer coefficient is determined by the following expression, which is established by empirical correlation for vertical plate and cylinders and horizontal plates (Kreith):

$$h_c = 0.0015\ \Delta T^{1/3}\ (\text{Btu/hr-in}^2\text{-°F})$$

where ΔT is the temperature difference between the cask surface and ambient.

Volumetric heat generation (Btu/hr-in^3) is applied to the active fuel region based on a total heat load of 12.5 kW, active fuel length of 91 inches and an axial power distribution, as shown in Figure 4.4.1.1-3.

Figure 4.4.1.5-1 Three-Dimensional Transfer Cask and Canister Model



4.4.1.6 Two-Dimensional Reconfigured Fuel Assembly Model

The two-dimensional reconfigured fuel assembly model is generated to calculate the temperature distribution of the hottest cross-section (1 inch long in the cask axial direction) of the Reconfigured Fuel Assembly (RFA). Because of symmetry, the model considers one-fourth of a cross-section of the RFA. The model is shown in Figure 4.4.1.6-1. ANSYS 'PLANE55' conduction elements and "LINK31" radiation elements are used in the model. The model includes a total of 16 fuel rods, 16 fuel tubes, the shell casing (the square tube with the same external dimensions as an intact fuel assembly) and the cover gas (considered to be Helium). Each fuel rod is located inside a stainless steel fuel tube. The fuel rod, which consists of the zircaloy clad, the fuel pellet (UO_2) and a small gap between the clad and fuel pellet, is modeled as a solid rod with the thermal conductivity of the UO_2. This is conservative, since the conductivity of UO_2 is less than that of the zircaloy and the main interest of the fuel rod is the cladding temperature. The gas between the fuel rod and the fuel tube, the gas between fuel tubes and the gas outside of the shell casing are considered to be helium.

As shown in Figure 4.4.1.6-1, radiation elements are defined between tubes and from tubes to the inner surface of the shell casing. Form factor of 1 is used for the radiation elements. Effective emissivity is computed using the following formula (Kreith) based on corresponding material emissivities:

$$\varepsilon_{eff} = 1/\ (1/\varepsilon_1 + 1/\varepsilon_2 -1)$$

where $\in_1$ and $\in_2$ are the emissivities of two parallel plates.

Radiation between the fuel rod and the fuel tube is conservatively ignored. Radiation between the shell casing and the inner surface of the fuel assembly tube is accounted for by establishing effective conductivities for the gas in the gap using the method described in Section 4.4.1.2.

Volumetric heat generation (Btu/hr-in^3) based on the design heat load of 0.0016 kW/pin is applied to the fuel rod elements. An active fuel length of 91 inches and a peaking factor of 1.15 are used.

Heat generation rate $= Q / V$

$= 0.6595$ Btu/hr-in^3

where:

Q = heat rate per pin (unit height)

$= (0.0016)\,(3413)\,(1.15)\,/(91) = 0.069$ Btu/hr

V = volume of pin (unit height)

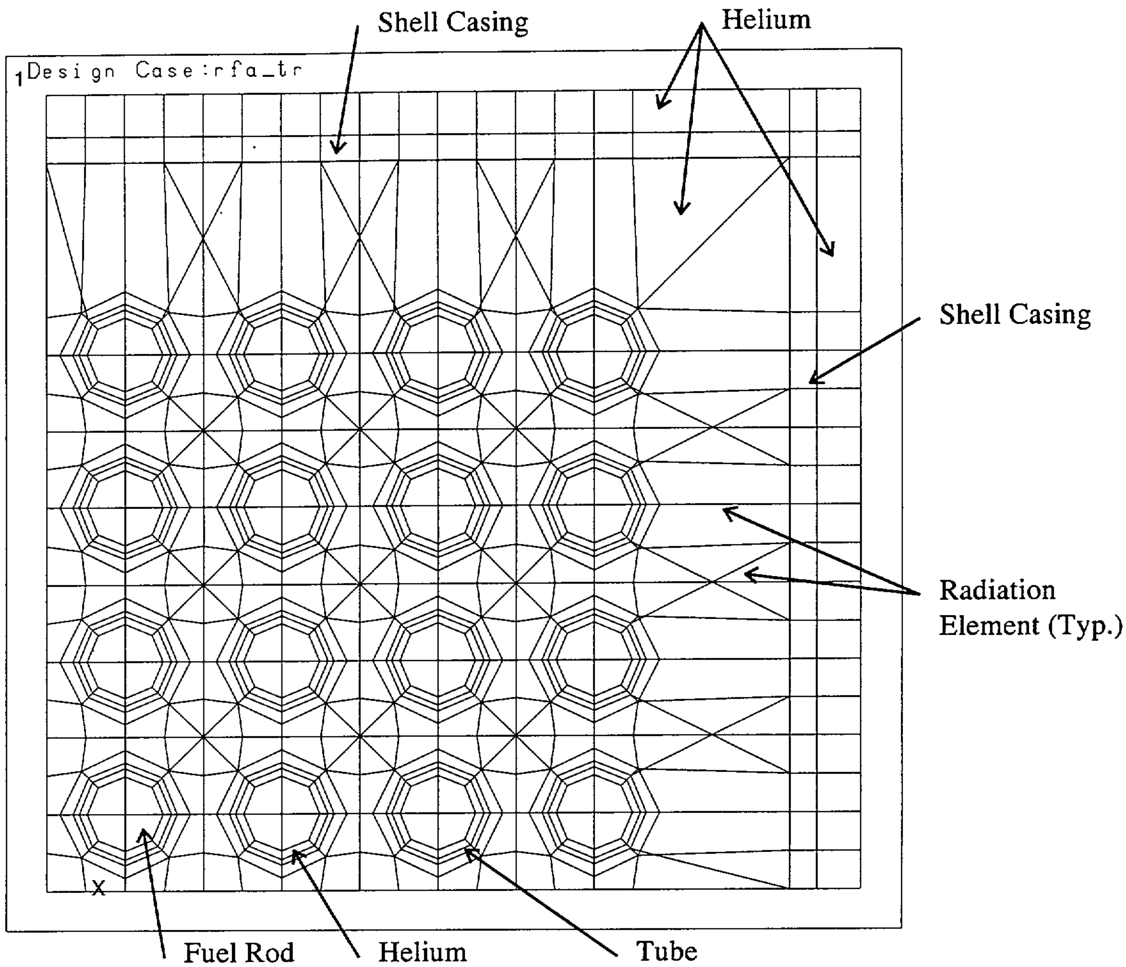
$= \pi\, 0.365^2\, /4 = 0.1046$ inch^3

Boundaries of the model at planes of symmetry (at X=0 and at Y=0) are considered to be adiabatic. The temperature at the right and top boundaries (at X=3.9 inch and at Y=3.9 inch) of the model is constrained to be uniform based on the maximum temperatures of the fuel tube as shown below. These temperatures envelope the calculated maximum tube temperatures for the design basis Yankee Class fuel assembly and are conservative, since the heat load for the RFA (0.102 kW) is less than one-third of the heat load for the design basis fuel assembly (0.347 kW).

Design Condition	Maximum Fuel Tube[1] Temperature (°F)
Normal	540
Off-Normal and Accident	580
Transfer	715

1. The Fuel Tube is shown in Drawing 455-881.

Figure 4.4.1.6-1 Two-Dimensional Reconfigured Fuel Assembly Model



4.4.2 Test Model

The NAC-MPC system is conservatively designed by analysis so that testing is not required.

4.4.3 Maximum Temperatures for Normal Conditions

Normal conditions consider, primarily, the temperature distribution in the spent fuel, the canister and the concrete cask in long-term storage. The routine operations of loading, closing and transfer of the canister to the concrete cask are transient conditions that result in different temperature conditions, depending on the configuration of the canister. In the transient conditions, the maximum temperature of the fuel is maintained below the maximum allowable short-term temperature limit (806°F) by implementing specific actions described in the procedures (Section 8.1.1) and in the Technical Specifications provided in Appendix 12A of Chapter 12. Temperature distributions for the off-normal and accident conditions are presented in Sections 11.1 and 11.2.

4.4.3.1 Maximum Temperatures in Long-Term Storage

Figure 4.4.3-1 shows the temperature distribution of the concrete cask and the canister for the normal, long-term storage condition. The air flow field and air temperatures in the annulus between the canister and the storage cask liner for the normal condition of storage are shown in Figures 4.4.3-2 and 4.4.3-3, respectively. The temperature distribution in the concrete portion of the storage cask is shown in Figure 4.4.3-4. The maximum component temperatures for the reconfigured fuel assembly are shown in Table 4.4.3-4.

As shown in Figure 4.4.3-3, a high-temperature gradient exists near the wall of the canister and at the liner of the concrete cask. The air in the center of the annulus exhibits a much lower temperature gradient, indicating cooler air. The higher temperature at the concrete cask steel liner surface indicates that radiation heat transfer occurs across the annulus. As shown in Figure 4.4.3-4, the local temperature in the concrete, directly affected by the radiation heat transfer across the annulus, can reach 165°F (347°K), which is less than the 200°F allowable temperature. The bulk temperature in the concrete, as determined by averaging the temperatures at the mid-radius of the concrete region, is less than the allowable value of 150°F.

4.4.3.2 Maximum Temperatures in Transient Operations

The time history of the maximum temperature of the fuel region in the canister for the transient conditions (the canister containing water for 20 hours, vacuum for 10 hours and helium for 36 hours while in the transfer cask) is shown in Figure 4.4.3-5.

Using the results for the temperature history evaluation presented in Figure 4.4.3-5, the heat up rate of the spent fuel is:

- 5.3°F per hour in the water condition
- 21.9°F per hour in the vacuum condition
- 4.8°F per hour in the helium condition

These heat up rates are used to develop the allowable time limits for completing required procedural steps for loading, closing and removing the canister from the transfer cask that are incorporated in the Technical Specifications provided in Chapter 12. The Technical Specifications specify the remedial actions required to ensure that the fuel cladding temperature does not exceed the maximum allowable short-term temperature limit of 806°F, if these time frames are not met. The actual loading, closing and transfer operations are expected to be completed in less time.

The fuel temperature at the start of the vacuum drying operation will be below 200°F based on the temperature of the water in the canister. The drying process is allowed to continue for 16 hours (LCO 3.1.5), which could result in a maximum fuel temperature of 550°F. The fuel is then placed in a helium condition for 26 hours (LCO 3.1.6), which could result in a fuel temperature of 675°F. The time limits imposed by the LCO thus ensure that the fuel cladding temperature does not exceed the maximum allowable short-term temperature of 806°F.

If the time limits of LCO 3.1.5 for the vacuum drying process are not met, the Technical Specifications require that the transfer cask and canister be returned to the spent fuel pool or that forced air cooling of the canister in the transfer cask be initiated. After 24 hours in either of these conditions, the fuel cladding temperature is expected to be below 466°F. This value is based on the maximum differential temperature between the fuel and the canister shell, as presented in Section 11.1.4.2, and a canister water temperature of 200°F. The vacuum drying process is then allowed to continue for 10 hours (LCO 3.1.5), which could result in a fuel temperature of 685°F. The canister may then be placed in a helium condition for up to 15 additional hours (LCO 3.1.6), which could result in a fuel temperature of 757°F. The time limits imposed by LCO 3.1.5 and LCO 3.1.6, thus ensure that the fuel cladding temperature does not exceed the maximum allowable short-term temperature limit of 806°F.

Consequently, for all of the transient conditions, the maximum fuel temperature, as controlled by the LCO time limits, is significantly below the maximum short-term temperature limit for the fuel of 806°F.

Figure 4.4.3-1 Temperature Distribution (°F) for Normal Storage

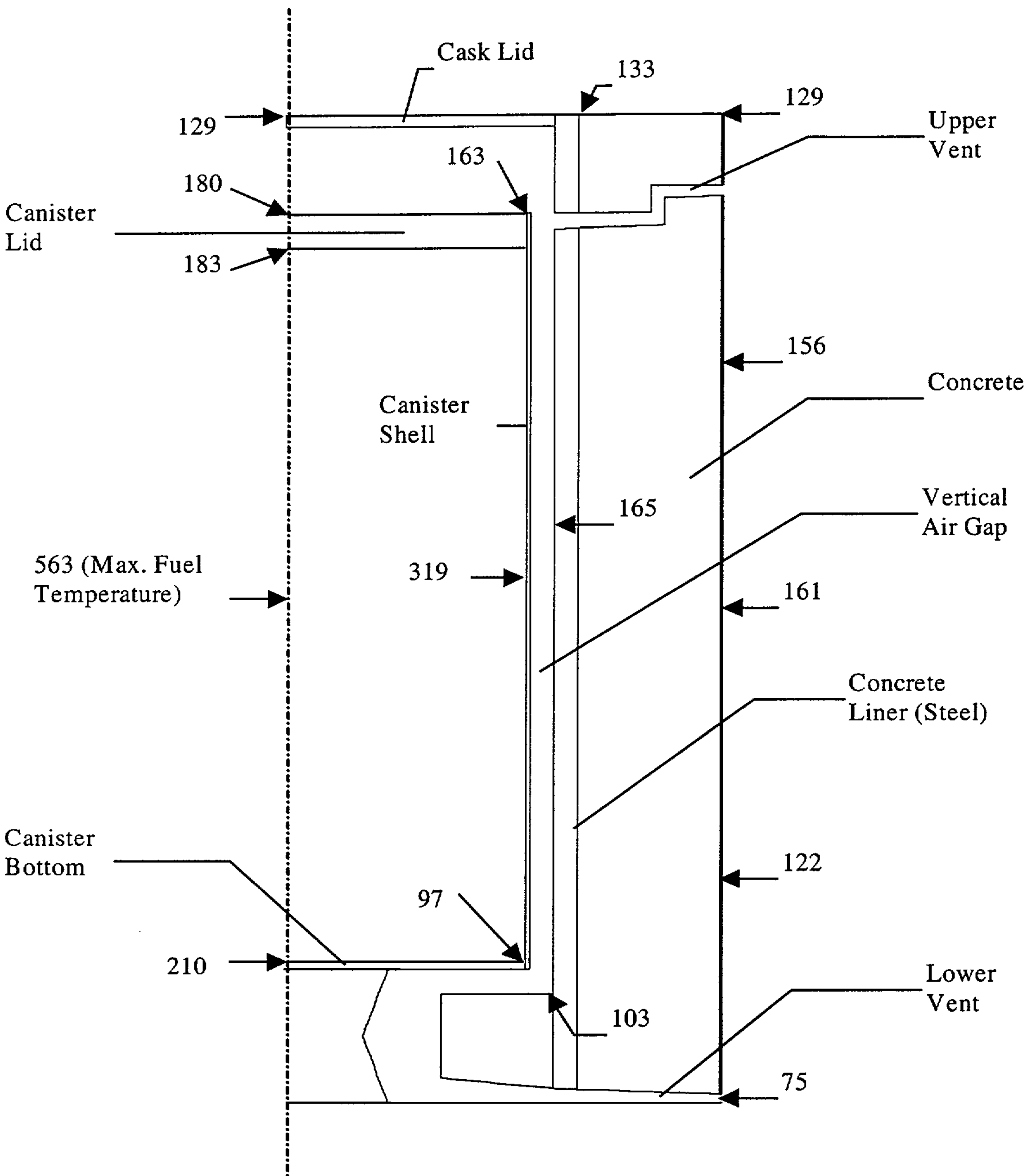


Figure 4.4.3-2 Air Flow Pattern in the Storage Cask in Normal Storage

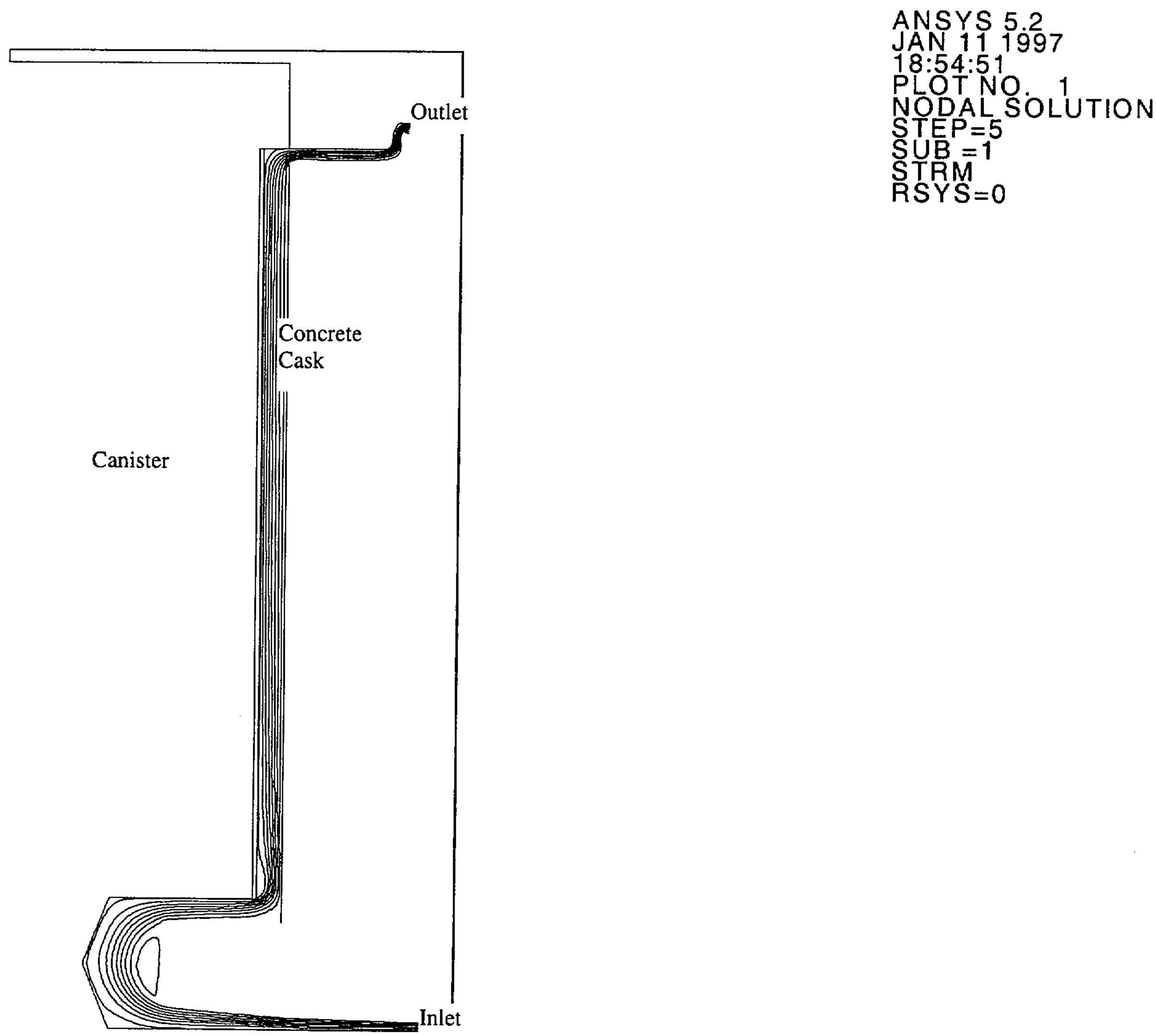
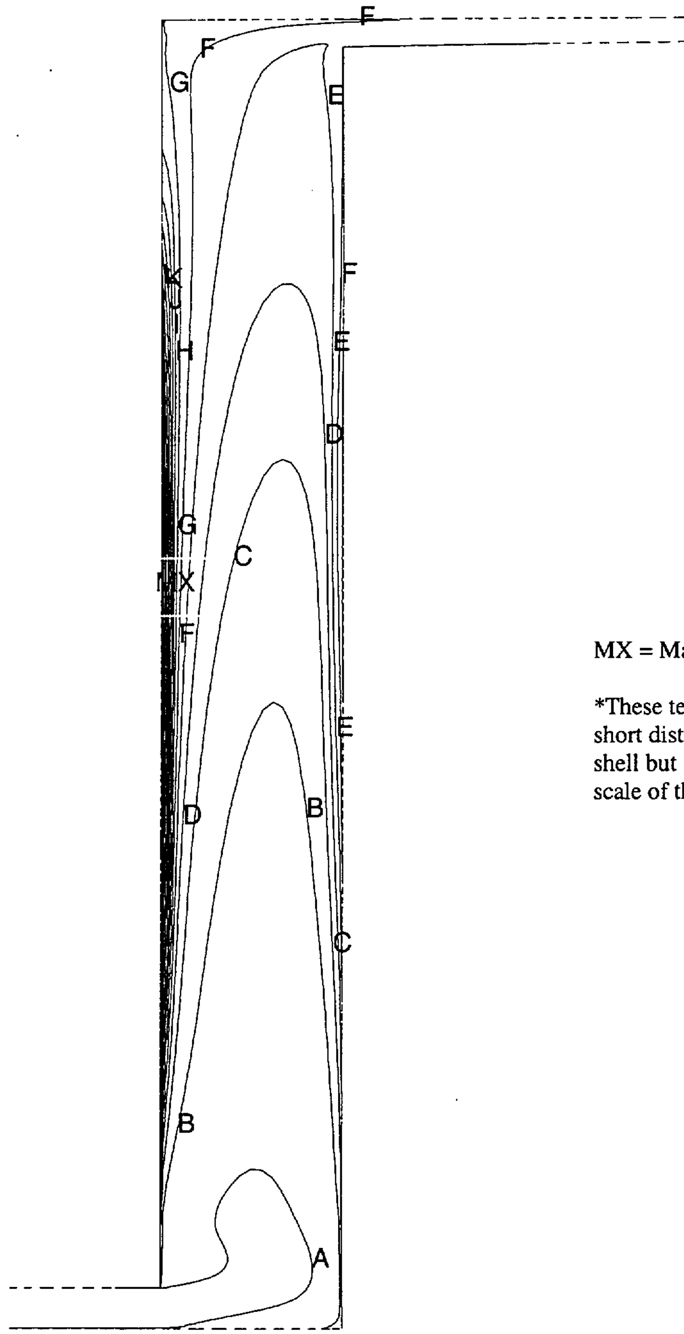


Figure 4.4.3-3 Air Temperature Field in the Storage Cask During the Normal Storage Condition

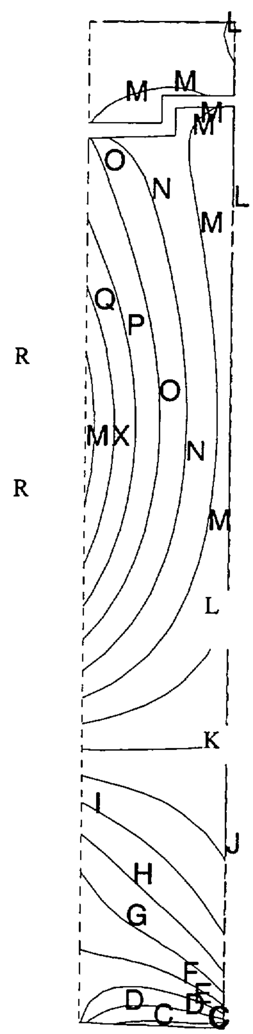


°F	
MX=	315
A=	79
B=	93
C=	106
D=	120
E=	133
F=	147
G=	160
H=	174
I=	187
J=	201
K=	214
L=	228
*M=	241
*N=	255
*O=	268
*P=	282
*Q=	295
*R=	309

MX = Maximum temperature

*These temperatures occur over short distances along the canister shell but are not shown due to the scale of the figure.

Figure 4.4.3-4 Concrete Temperature Field During the Normal Storage Condition



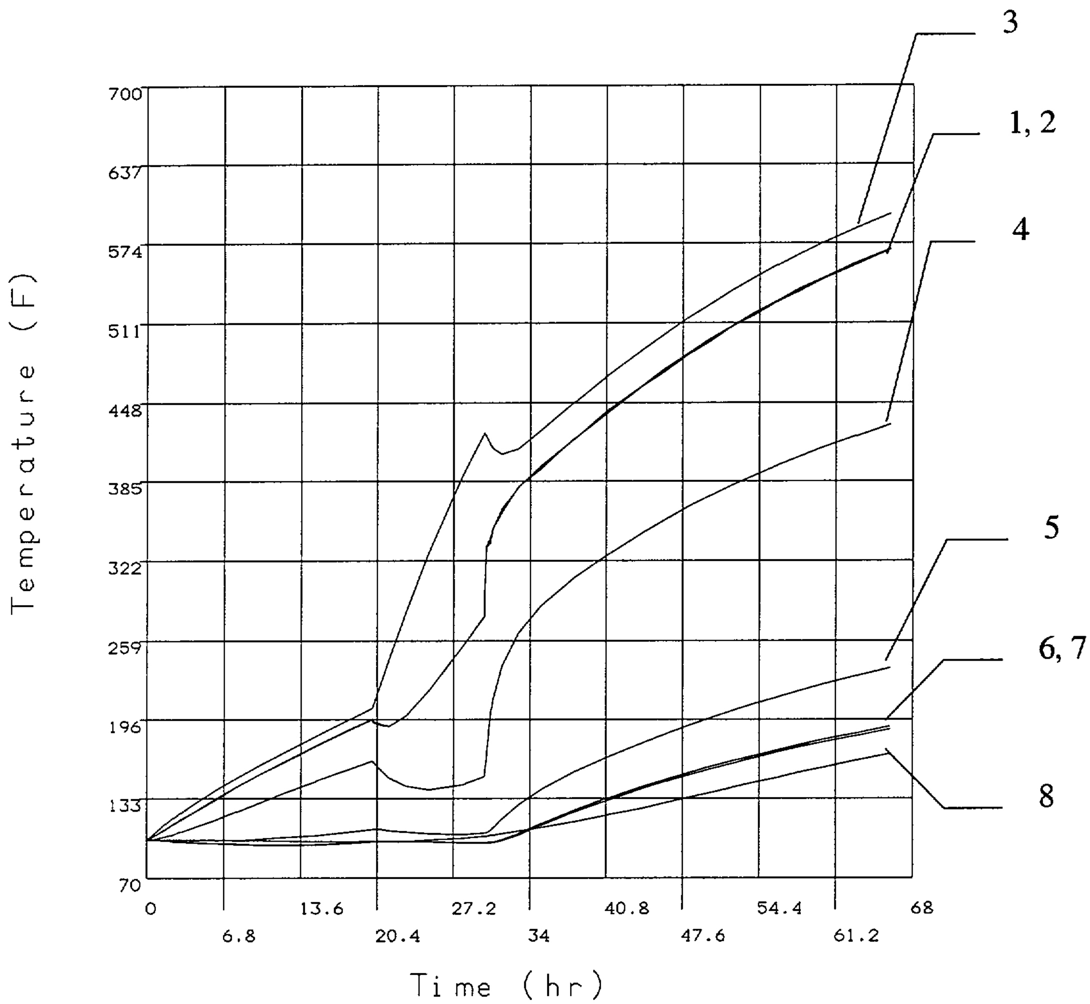
°F

MX= 165
*A= 77
*B= 82
C= 87
D= 92
E= 97
F= 102
G= 107
H= 112
I= 117
J= 122
K= 127
L= 132
M= 137
N= 142
O= 147
P= 152
Q= 157
R= 162

MX = Maximum temperature

*These temperatures occur in the lower edge of the concrete above the inlet vent. They are not shown due to the scale of the figure.

Figure 4.4.3-5 History of Maximum Component Temperatures for the Nominal Transfer Conditions



1. Temperature of Support Disks
2. Temperature of Aluminum Disks
3. Temperature of the Fuel
4. Temperature of the Canister
5. Temperature of the Inner Shell
6. Temperature of the Lead
7. Temperature of the Neutron Shield
8. Temperature of Transfer Cask Doors

Table 4.4.3-1 Maximum Component Temperatures for the Normal Condition of Storage

Component	**Maximum Temperature (°F)**	**Allowable Temperature (°F)**
Fuel Cladding	563	644
Aluminum Disk	527	650
Support Disk	529	650
Canister	319	800
Concrete Liner (steel)	165	700
Concrete	165 (local)	200 (local)
	133 (bulk*)	150 (bulk)

* The average temperature of the concrete region is used as the bulk concrete temperature.

Table 4.4.3-2 Maximum Component Temperatures for the Nominal Helium Transfer Condition

Component	Maximum Temperature (°F)[1]	Allowable Temperature (°F)
Fuel	597	806
Lead	191	600
Neutron Shield	188	300
Aluminum Disk	569	700
Support Disk	570	800
Canister	430	800
Transfer Cask Shells	237	700

1. Maximum temperatures calculated for the nominal transient condition that considers the canister inside of the transfer cask containing water for 20 hours, a vacuum for 10 hours, and helium for 36 hours.

Table 4.4.3-3 Maximum Component Temperatures for the Vacuum Transfer Condition

Component	Maximum Temperature (°F)[1]	Allowable Temperature (°F)
Fuel	424	806
Lead	98	600
Neutron Shield	98	300
Aluminum Disk	278	700
Support Disk	279	800
Canister	151	800
Transfer Cask Shells	105	700

1. Maximum temperatures calculated for the transient condition that considers the canister inside of the transfer cask containing water for 20 hours and a vacuum for 10 hours.

Table 4.4.3-4 Maximum Component Temperatures for the Reconfigured Fuel Assembly

Design Condition	**Maximum Temperatures (°F)**			
	PWR Fuel Tube[1,2]	**Shell Casing**[3,4]	**Reconfigured Fuel Assembly Tube**[5,6]	**Fuel Rod Cladding**[7]
Normal	540	543	563	563
Off-Normal and Accident Conditions	580	583	602	602
Transfer	715	718	734	734

1. Fuel Tube as shown in Drawing 455-881.
2. Bounding fuel tube temperatures as described in Section 4.4.1.6.
3. Reconfigured Fuel Assembly Shell Casing as shown in Figure 4.4.1.6-1.
4. Material allowable temperature: 800°F.
5. Reconfigured Fuel Assembly (Fuel) Tube as shown in Figure 4.4.1.6-1.
6. Material allowable temperature: 800°F.
7. Fuel cladding allowable temperatures:

   Normal conditions: 644°F

   Off-normal, Accident and Transfer conditions: 806°F.

4.4.4 Minimum Temperatures

Section 11.1 provides the temperature distribution for the off-normal severe cold environmental conditions of -40°F. At this extreme condition, the components are above their minimum material limits.

4.4.5 Maximum Internal Pressure for Normal Conditions

The NAC-MPC canister is backfilled with helium to atmospheric pressure (0.0 psig) and closed by welding. Normal condition pressure comprises the pressure, due to the heating of the backfilled helium, plus the pressure due to the postulated failure of 3 percent of the stored fuel rods with the subsequent release of 30 percent of the fission gas and all of the rod charge gas to the canister cavity, at temperature, from those failed rods. All of the gases except the fission gases are assumed to be helium. The total pressure for each volume is found by calculating the molar quantity of each gas and summing those directly. The calculated average temperature of the helium gas is 442°F based on the thermal analysis results using the three-dimensional canister model described in Section 4.4.1.2. The pressure is calculated using the Ideal Gas Law and applying a conservative average temperature of 450°F. The gas constant, R, is 0.0821 (atm x liters)/(Mole °K) (Lamarsh). The design basis fuel assembly for the internal pressure calculation is the CE Type A assembly. This assembly has the highest rod backfill pressure (315 psig) and received the highest burnup (36,000 MWd/MTU).

The number of moles of the backfill gases is calculated using the Ideal Gas Law, PV = NRT. Backfill gas for the canister is assumed to be initially at 1 atmosphere absolute. The quantity of fission gas is derived from the SAS2H generated isotopics of the CE Type A fuel assembly. The release of fission gas is as assumed for directly loaded fuel. For normal operating conditions, 3 percent of the fuel rods are assumed to fail, releasing 30 percent of their total fission gas and all of the backfill helium.

The fuel rod plenum volume is:

$$V_1 = \pi r^2 L - \frac{M_{Spring}}{r}$$

$$V_1 = \pi\left\{\left(\frac{0.317 \text{ inches}}{2}\right)^2 \times 1.942 \text{ inches}\right\} - \frac{\left(3.3 \text{ g} \times 2.2046 \times 10^{-3} \frac{\text{lb}}{\text{g}}\right)}{0.288 \frac{\text{lb}}{\text{inch}^3}} = 0.1280 \text{ inches}^3$$

The pellet clad gap volume is:

$$V_2 = \pi L\left(r^2_{\text{Clad ID}} - r^2_{\text{Pellet OD}}\right)$$

$$V_3 = \pi \times r^2_{\text{Clad ID}} \times L$$

$$V_3 = \pi \times \frac{(0.317 \text{ inches})^2}{4} \times 2.458 \text{ inches} = 0.1940 \text{ inches}^3$$

The total fuel rod backfill volume is:

$$V_{\text{Rod Back-Fill}} = V_1 + V_2 + V_3$$

$$V_{\text{Rod Back-Fill}} = 0.1280 \text{ inches}^3 + 0.2915 \text{ inches}^3 + 0.1940 \text{ inches}^3 = 0.6135 \text{ inches}^3$$

For the loaded canister, the total backfill gas volume is:

$$V = \text{Total Back - Fill } 0.6135 \text{ inches}^3 \times 231 \frac{\text{Rods}}{\text{Assembly}} \times 36 \frac{\text{Assemblies}}{\text{Cask}} \times \left(2.54 \frac{\text{cm}}{\text{inch}}\right)^3 \times \frac{0.001\ \ell}{\text{cm}^3}$$

$$= 83.605 \frac{\ell}{\text{Cask}}$$

From the rod backfill volume and pressure, the quantity of rod backfill gas is calculated using the ideal gas law:

$$N = \frac{Pv}{RT}$$

$$N = \frac{\left\{(315\ \text{psig} + 14.7) \times \dfrac{1\ \text{atm}}{14.7\ \text{psia}}\right\} \times 83.605\ \dfrac{\ell}{\text{Cask}}}{0.0821\ \dfrac{\text{atm}\ \ell}{\text{Mole K}} \times 293\ \text{K}} = 77.95\ \frac{\text{Moles of Rod Fill Gas}}{\text{Cask}}$$

The number of moles of fission gas per assembly is:

Isotope	Atomic Weight (gram/mole)	Mass (gram)	Number of Moles
KR	83	10.9	0.131
KR	84	29.9	0.356
KR	85	3.54	0.042
KR	86	48.4	0.563
I	127	11	0.087
I	129	47.2	0.366
XE	130	1.78	0.014
XE	131	112	0.855
XE	132	285	2.159
XE	134	395	2.948
XE	136	577	4.243
Total	---	---	11.76

There is a maximum of 36 assemblies per cask. Therefore the number of moles of fission gas per cask is 36 times that of the single assembly.

$$N = 36\frac{\text{Assemblies}}{\text{Cask}} \times 11.76\frac{\text{Moles}}{\text{Assembly}} = 423.44\frac{\text{Moles of Fission Gas}}{\text{Cask}}$$

The canister is backfilled to 1.5 atmosphere with helium at room temperature for leak testing, after which the pressure is reduced to 1 atmosphere. During leak testing the temperature of the helium may rise above the 150°F assumed in the pressure evaluation.

$$V^{TSC}_{\text{Free Gas Volume}} = V_{\text{Canister}} - \left(\frac{\left(M_{\text{TSC Shield Lid}} + M_{\text{TSC Structural Lid}}\right)}{\rho_{\text{Steel}}} + V^{TSC}_{\text{Basket}} + V_{\text{Fuel}} \right)$$

$$V_{\text{Canister}} = \pi \frac{d^2}{4} \left(L_{\text{Canister}} - L_{\text{TSC Bottom Plate}}\right) = \pi \times \frac{(69.39 \text{ inches})^2}{4} \times (122.50 \text{ inches} - 1.0 \text{ inch})$$

$$V_{\text{Canister}} = 459{,}472.93 \text{ inches}^3$$

$$V^{TSC}_{\text{Basket}} = \frac{\left(M^{TSC}_{\text{Basket}} - \left(M_{\text{BORAL}} + M_{\text{Aluminum}} + M_{\text{Support Disks}}\right)\right)}{\rho_{\text{Steel}}} + \frac{M_{\text{BORAL}}}{\rho_{\text{BORAL}}} + \frac{M_{\text{Aluminum}}}{\rho_{\text{Aluminum}}} + \frac{M_{\text{Support Disk}}}{\rho_{\text{17-4-PH}}}$$

$$V^{TSC}_{\text{Basket}} = \frac{\left(9{,}530 \text{ lb} - \left(694.81 \text{ lb} + 810 \text{ lb} + 3{,}720 \text{ lb}\right)\right)}{0.288 \frac{\text{lb}}{\text{in}^3}} + \frac{694.81 \text{ lb}}{0.095 \frac{\text{lb}}{\text{in}^3}} + \frac{810 \text{ lb}}{0.098 \frac{\text{lb}}{\text{in}^3}} + \frac{3{,}720 \text{ lb}}{0.282 \frac{\text{lb}}{\text{in}^3}}$$

$$V^{TSC}_{\text{Basket}} = 43{,}719.16 \text{ inches}^3$$

$$V^{TSC}_{\text{Free Gas Volume}} = 459{,}472.93 - \left(\frac{\left(5{,}390 \text{ lb} + 3{,}230 \text{ lb}\right)}{0.288 \frac{\text{lb}}{\text{in}^3}} + 43{,}719.16 \text{ inches}^3 + 88{,}171.78 \text{ inches}^3 \right)$$

$$V^{TSC}_{\text{Free Gas Volume}} = 297{,}651.43 \frac{\text{inches}^3}{\text{Cask}}$$

$$V^{TSC}_{\text{Free Gas Volume}} = 297{,}651.43 \frac{\text{inches}^3}{\text{Cask}} \times \frac{1 \ \ell}{61.02 \text{ inches}^3} = 4{,}877.93 \frac{\ell}{\text{Cask}}$$

$$N = \frac{1 \text{ atm} \times 4{,}877.93 \frac{\ell}{\text{Cask}}}{0.0821 \frac{\text{atm } \ell}{\text{Mole K}} \times 339 \text{ K}} = 175.26 \frac{\text{Moles of TSC Backfill Gas}}{\text{Cask}}$$

The maximum normal operating pressure (MNOP) in the canister is calculated using the ideal gas law where:

$$N = N_{\text{TSC Back-Fill}} + 0.03\left(N_{\text{Rod Back-Fill}}\right) + 0.3(0.03)\left(N_{\text{Fission Gas}}\right)$$

$$N = 175.26\,\frac{\text{Moles}}{\text{Cask}} + 0.03\left(77.95\,\frac{\text{Moles}}{\text{Cask}}\right) + 0.3(0.03)\left(423.44\,\frac{\text{Moles}}{\text{Cask}}\right)$$

$$N = 181.41\,\frac{\text{Moles}}{\text{Cask}}$$

Therefore, the maximum normal operating condition canister internal pressure is:

$$P = \frac{\left(181.41\,\frac{\text{Moles}}{\text{Cask}}\right)\times\left(0.0821\,\frac{\text{atm}\,\ell}{\text{mole K}}\right)\times 505.37\text{ K}}{\left(4{,}877.93\,\frac{\ell}{\text{Cask}}\right)} = 1.54\text{ atm} \approx 22.6\text{ psia} \approx 7.9\text{ psig}$$

4.4.6 Maximum Thermal Stresses for Normal Conditions

The canister and concrete storage cask thermal stresses are evaluated in Section 3.4.4.

4.4.7 Evaluation of Cask Performance for Normal Conditions of Storage

As shown in the preceding sections, the NAC-MPC system operates within the thermal design limits. Therefore, no degradation due to temperature effects on material or components is expected over the lifetime of the cask.

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5.0 SHIELDING EVALUATION

5.1 Discussion and Results

The regulation governing spent fuel storage, 10 CFR 72, does not establish specific cask dose rate limits. However, 10 CFR 72.104 and 10 CFR 72.106 specify that for an array of casks in an Independent Spent Fuel Storage Installation (ISFSI), the annual dose to an individual outside the controlled area boundary must not exceed 25 mrem to the whole body, 75 mrem to the thyroid and 25 mrem to any other organ during normal operations. In the case of a design basis accident, the dose to an individual outside the area boundary must not exceed 5 rem to the whole body or any organ. The ISFSI must be at least 100 meters from the owner controlled area boundary. In addition, the occupational dose limits and radiation dose limits for individual members of the public in 10 CFR Part 20 (Subparts C and D) must be met. Chapter 10, Section 10.3, demonstrates NAC-MPC compliance with the requirements of 10 CFR 72 with regard to annual and occupational doses at the owner controlled area boundary. This chapter presents the shielding evaluations of the NAC-MPC storage system. Dose rate profiles are calculated as a function of distance from the side, top and bottom of the NAC-MPC storage and transfer casks. Shielded source terms from the NAC-MPC storage cask are calculated to establish owner controlled area boundary dose estimates due to the presence of the ISFSI.

The NAC-MPC accommodates up to 36 CE Yankee Class fuel assemblies with a maximum of 36,000 MWD/MTU burnup and with a minimum of 8 years cool time. While 8.1 years cooling is required to meet cask total heat load requirements, 8.0 years is conservatively used as the shielding design basis. CE fuel with this burnup and cool time is defined as the design basis fuel. CE, UN and Westinghouse Yankee Class fuel assemblies with a maximum burnup of 32,000 MWD/MTU at 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 and 16 years for assemblies containing Zircaloy and stainless steel fuel region hardware, respectively. For shielding evaluation purposes the Exxon assembly type is identical to the CE fuel. The physical parameters of the Yankee Class fuel assemblies are presented in Table 5.2-1.

The NAC-MPC storage system is comprised of a transportable storage canister, a transfer cask, and a vertical concrete storage cask. License drawings for these items are provided in Section 1.5. The transfer cask containing the canister and the basket is loaded under water in the spent fuel pool. Once filled with fuel, the shield lid is placed on top of the canister and transfer cask is removed from the pool. After draining about 12 inches of water (approximately 50 gallons) from the cavity, the shield lid is welded in place, and the canister is drained and dried. Finally, the structural lid is welded in place. The transfer cask is then used to transfer the canister to the storage cask where it is stored dry until transport. Shielding evaluations are performed for the transfer cask with both a wet and dry canister cavity as would occur during the welding of the shield lid and during the welding of the structural lid, respectively. Shielding evaluations are performed for the storage cask with the cavity dry.

A canister may contain one or more reconfigured fuel assemblies. The reconfigured fuel assembly is designed to confine Yankee Class spent fuel rods, or portions thereof, which have been classified as failed. Each assembly can accommodate up to a total of 64 fuel pins, which is significantly less than other Yankee Class fuel assemblies. A depiction of the assembly is provided in Figure 1.2-5. Because the source term (neutron and gamma) is directly proportional to fuel mass, for a given burnup and enrichment, the reconfigured assembly source term is bounded by that of a design basis fuel assembly. Consequently, a separate shielding analysis is not required for the reconfigured fuel assembly.

The transfer cask has a multiwall radial shield comprised of 0.75 inches of carbon steel, 3.5 inches of lead, 2 inches of solid borated polymer (NS-4-FR), and 1.25 inches of carbon steel. An additional 0.625 inch of stainless steel shielding is provided, radially, by the canister shell. Gamma shielding is provided primarily by the steel and lead layers, and neutron shielding is provided primarily by the NS-4-FR. The transfer cask bottom shield design is a solid section of 9.50 inches of carbon steel. The top shielding is provided by the stainless steel canister shield and structural lids, which are 5 inches and 3 inches thick, respectively. In addition, 5 inches of carbon steel is used as temporary shielding during welding, draining, drying and helium backfill operations. This temporary shielding is removed prior to storage.

The storage cask radial shield design is comprised of a 3.5-inch thick carbon steel inner liner surrounded by 21 inches of concrete. Gamma shielding is provided by both the carbon steel and concrete, and neutron shielding is provided primarily by the concrete. As in the transfer cask, an additional 0.625 inch thickness of stainless steel radial gamma shielding is provided by the canister shell. The storage cask top shielding design is comprised of 8 inches of stainless steel from the canister lids, a shield plug containing a 1 inch thickness of NS-4-FR and 4.125 inches of carbon steel, and a 1.5 inch thick carbon steel lid. Since the bottom of the storage cask sits on a concrete pad, the storage cask bottom shielding is comprised of 1 inch of stainless steel from the canister bottom plate, 2 inches of carbon steel (pedestal plate) and 1 inch of carbon steel cask base plate. The base plate and pedestal base are structural components that position the canister above the air inlets. The cask base plate supports the storage cask during lifting, and forms the cooling air inlet channels at the cask bottom.

Shielding evaluations of the NAC-MPC transfer and storage casks are performed with SCALE 4.3 for the PC (ORNL). In particular, the SCALE shielding analysis sequence SAS2H (Herman) is used to generate source terms for the design basis fuel, using the 27 group ENDF/B-IV (Jordan) library, 27GROUPNDF4. SAS1 (Knight) is used to perform one-dimensional radial and axial shielding analysis, and a modified version of SAS4 (Tang) is used to perform three-dimensional shielding analysis. The SCALE 4/3 SAS4 code sequence has been modified to allow multiple surface detectors; the new code sequence is entitled SAS4A. The use of the surface subdetectors enables the user to obtain surface profiles of the detector response and the surface tallies on the cask surfaces other than the cask shield. Dose tally routines were modified to accept user-defined surface detectors instead of the fixed surfaces, 1m, 2m, and 4m detectors in SCALE 4.3 SAS4. Each surface can be defined by specifying the cylindrical or disk surface tally location and extent. Each surface may be broken into multiple subdetectors. Code modifications were tested against the SCALE 4.3 manual and NAC test cases. Reliability of the subdetectors was verified by comparison to point detector results. The 27 group neutron, 18 group gamma, coupled cross section library (27N-18COUPLE) based on ENDF/B-IV is used in all shielding evaluations. Source terms include: fuel neutron, fuel gamma, and activated hardware gamma. Dose rate evaluations include the effect of fuel burnup peaking on fuel neutron and gamma source terms.

Dose rate profiles are shown for the storage and transfer casks in Section 5.4.3. Maximum dose rates for the storage cask under normal and accident conditions are shown in Table 5.1-1 for design basis fuel. These dose rates are based on three-dimensional Monte Carlo and one-dimensional discrete ordinates calculations. Monte Carlo error (1σ) is indicated in parenthesis. In normal conditions with design basis fuel, the storage cask maximum side dose rate is 47.3 (0.4%) mrem/hr at the bottom endfitting elevation and 54.0 (4.9%) mrem/hr on the top lid surface just above the heat transfer annulus. Since the storage cask is vertical during normal storage operation, the bottom is inaccessible. The dose rates at the inlet and outlet vents are 99.0 (5.4%) mrem/hr and 23.8 mrem/hr (5.0%) due to radiation streaming. Under accident conditions involving a projectile impact and a loss of 6 inches of concrete, the surface dose rate increases to 314 mrem/hr with design basis fuel. There are no design basis accidents that result in a tip-over of the NAC-MPC storage cask.

Maximum dose rates for the transfer cask with design basis fuel and with a wet and dry canister cavity are shown in Table 5.1-2. The maximum dose rates with design basis fuel and the canister cavity wet during shield lid welding operations are 210.2 (0.8%), 188.7 (1.1%) and 77.2 (0.7%)

mrem/hr on the side, top, and bottom, respectively. The maximum dose rates with design basis fuel and the canister cavity dry during structural lid welding operations are 413.4 (1.5%), 358.9 (2.6%) and 398.0 (3.9%) mrem/hr on the side, top, and bottom, respectively. These values include the addition of 5 inches of carbon steel operational shielding installed on the shield lid during its closure and on the structural lid during its handling and closure. In normal operations during welding of the canister lids, the bottom of the transfer cask is generally inaccessible.

Table 5.1-1 Summary of NAC-MPC Storage Cask Maximum Dose Rates with Design Basis Fuel

Condition	Source	Cask Surface (mrem/hr)			1 Meter From Surface (mrem/hr)		
		Side	Top	Bottom[1]	Side	Top	Bottom[1]
NORMAL	neutron	0.3	42.5	32.0	0.3	11.4	2.3
	gamma	47.0	11.5	67.0	21.5	4.0	2.8
	Total	47.3 (0.4%)	54.0 (4.9%)	99.0 (5.4%)	21.8 (1.3%)	15.4 (2.5%)	5.1 (6.0%)
POSTULATED ACCIDENT[2]	neutron	4.0	na	na	1.5	na	na
	gamma	310.0	na	na	137.5	na	na
	Total	314.0	na	na	139.0	na	na

Table 5.1-2 Summary of NAC-MPC Transfer Cask Maximum Dose Rates with Design Basis Fuel

Condition	Source	Cask Surface (mrem/hr)			1 Meter From Surface (mrem/hr)		
		Side	Top	Bottom	Side	Top	Bottom
NORMAL WET[3]	neutron	0.0	1.2	0.2	0.7	0.3	0.0
	gamma	210.2	187.5	77.0	39.8	388.8	19.0
	Total	210.2 (0.8%)	188.7 (1.1%)	77.2 (0.7%)	40.5 (0.7%)	389.1 (5.1%)	19.0 (2.1%)
NORMAL DRY[4]	neutron	77.5	116.6	276.2	44.8	13.5	33.4
	gamma	335.9	242.3	121.8	58.6	26.1	33.3
	Total	413.4 (1.5%)	358.9 (2.6%)	398.0 (3.9%)	103.4 (0.6%)	39.6 (4.5%)	66.7 (3.6 %)

[1] Bottom surface is inaccessible. Dose rates adjacent to bottom inlet indicated.
[2] Projectile impact, 6 inches loss of concrete.
[3] 5 inches of carbon steel temporary shielding, shield lid in position.
[4] 5 inches of carbon steel temporary shielding, shield lid and structural lid in position.

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5.2 Source Specification

The NAC-MPC storage system can safely transfer and store Yankee Class fuel from four vendors in two fuel rod configurations. These are: Combustion Engineering (CE) 16 x 16 Type A and Type B, Exxon 16 x 16 Type A and Type B, United Nuclear (UN) 16 x 16 Type A and Type B, and Westinghouse (WE) 18 x 18 stainless steel clad Type A and Type B. The geometry of the Type A and Type B fuel assemblies allows a cruciform control rod to be inserted between assemblies during reactor operation. The cross-section of a typical Yankee Class fuel assembly is shown in Figure 5.2-1.

The NAC-MPC accommodates up to 36 CE Yankee Class fuel assemblies with a maximum of 36,000 MWD/MTU burnup and with a minimum of 8 years cool time. While 8.1 years cooling is required to meet cask total heat load requirements, 8.0 years is conservatively used as the shielding design basis. CE fuel with this burnup and cool time is defined as the design basis fuel. CE, UN and Westinghouse Yankee Class fuel assemblies with a maximum burnup of 32,000 MWD/MTU at 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 and 16 years for assemblies containing Zircaloy and stainless steel fuel region hardware, respectively. For shielding evaluation purposes, the Exxon assembly type is identical to the CE fuel. The physical parameters of the Yankee Class fuel assemblies are presented in Table 5.2-1.

The SAS2H code sequence (Herman) is used to generate source terms. This code sequence is part of the SCALE 4.3 code package for the PC (ORNL). SAS2H includes an XSDRNPM (Greene) neutronics model of the fuel assembly and ORIGEN-S (Herman) fuel depletion/source terms calculations. The 27 energy group ENDF/B-IV neutron cross section library, 27GROUPNDF4, is used in the source terms calculations. Source terms are generated for both UO_2 fuel and fuel assembly hardware. The hardware activation is calculated by light element transmutation using the incore neutron flux spectrum produced by the SAS2H neutronics model. The hardware is assumed to be Type 304 stainless steel with 1.2 g/kg of ^{59}Co impurity. The effects of axial flux spectrum and magnitude variation on hardware activation are estimated by flux ratios based on empirical data (Luskic).

An evaluation of the Yankee Class fuel types established the Combustion Engineering (CE) 16 x 16 Type A fuel assembly (Table 5.2-2) at 36,000 MWD/MTU burnup and 8 years cool time as the Yankee Class design basis fuel assembly for the shielding evaluations. A minimum fuel enrichment of 3.7 wt % ^{235}U is assumed to maximize the fuel neutron source. Reactor operating conditions assumed for the analysis are shown in Table 5.2-3.

The NAC-MPC design basis fuel source terms are shown Table 5.2-4. Source strengths are defined for five source regions: active fuel, upper end fitting, upper plenum, lower end fitting and lower plenum. The fuel assembly length, active fuel region length and fuel assembly hardware elevations are shown for the design basis fuel assembly in Figure 5.2-2.

5.2.1 Gamma Source

The design basis fuel and hardware gamma source spectra are shown in Table 5.2-5. The fuel gamma source contains contributions from both fission products and actinides. The spectra are presented in the 18 group structure consistent with the SCALE 4.3 27N-18COUPLE cross section library. The hardware gamma spectra contains contributions primarily from ^{60}Co due to the activation of Type 304 stainless steel with 1.2 g/kg ^{59}Co impurity and with some minor contributions from ^{59}Ni and ^{58}Fe. The magnitude of this spectra is based on the irradiation of 1 kg of stainless steel in the incore flux spectrum produced by the SAS2H neutronics calculation.

The activated fuel assembly hardware source terms are found by multiplying the source strength from 1 kilogram by the number of kilograms of steel or inconel material in the plenum, upper end fitting and lower end fitting regions, and by multiplying by a regional flux ratio. The regional flux ratio accounts for the effects of both magnitude and spectrum variation on hardware activation. These ratios are based on empirical data (Luskic). A flux ratio of 0.2 is applied to hardware regions directly adjacent to the active core region, i.e., upper and lower plenum. A flux ratio of 0.1 is applied to hardware regions once removed from the active core region, i.e., upper and lower end fitting region.

5.2.2 Neutron Source

The design basis fuel neutron spectrum is shown in Table 5.2-6. The neutron source results from actinide spontaneous fission and from (α,n) reactions with the oxygen in UO_2. The isotopes ^{242}Cm and ^{244}Cm characteristically produce all, but a few percent of the spontaneous fission neutrons and (α,n) source in light water reactor fuel. The next largest contribution is from (α,n) reactions from ^{238}Pu. The neutron spectra from spontaneous fission are based on fission spectrum measurements of ^{235}U and ^{252}Cf. Neutron spectra from (α,n) reactions is based on Po-α-O source measurements. These spectra are included in the ORIGEN-S nuclear data libraries of the SCALE 4.3 code package. The spectra are automatically collapsed from the energy group structure of the data library into that of the SCALE 27 group neutron cross section library (Herman).

5.2.3 Source Axial Profile

An enveloping burnup shape for three-dimensional shielding and thermal evaluations is created based on core depletion calculations for the Yankee Class fuel. The normalized burnup profile, averaged over the range from 30,000 to 36,000 MWD/MTU, and the corresponding enveloping shape, is shown in Figure 5.2-3. A burnup peak of 1.15 is found to envelope the design basis fuel axial burnup distribution. The corresponding gamma and neutron source distribution is shown in Figure 5.2-4. The gamma source distribution follows the burnup shape directly and has a 1.15 peaking factor. However, the neutron source distribution peaks to a higher level. Based on SAS2H calculations of the neutron source magnitude as a function of burnup, a 4.2 power dependence of the neutron source on burnup, i.e., neutron source ~ $B^{4.2}$, is exhibited. This yields a 1.80 peaking factor for neutrons.

Figure 5.2-1 Yankee Class Combustion Engineering Design Basis Fuel Assembly Array

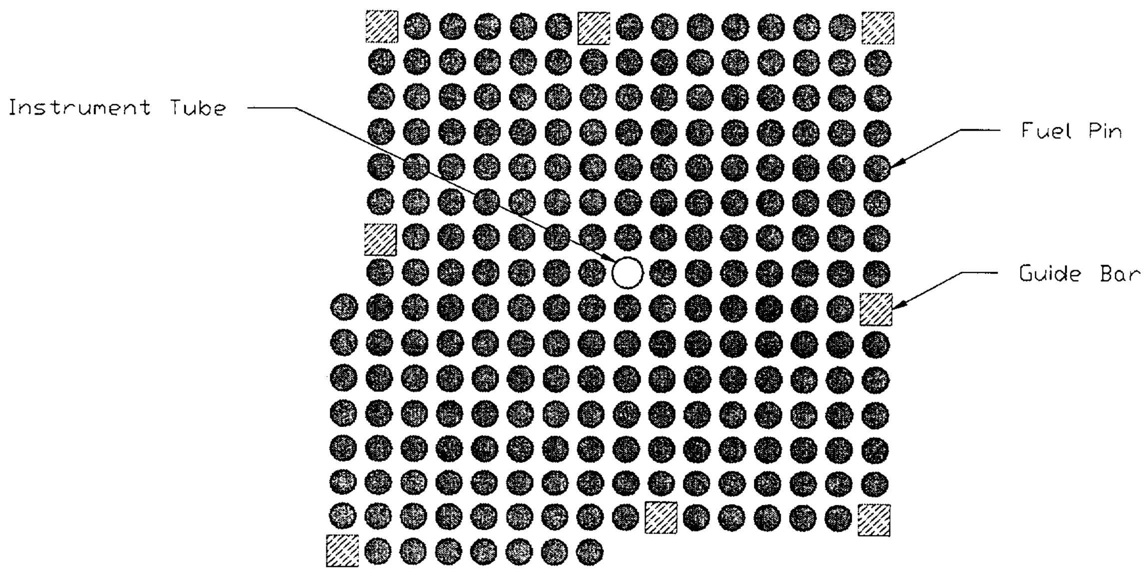
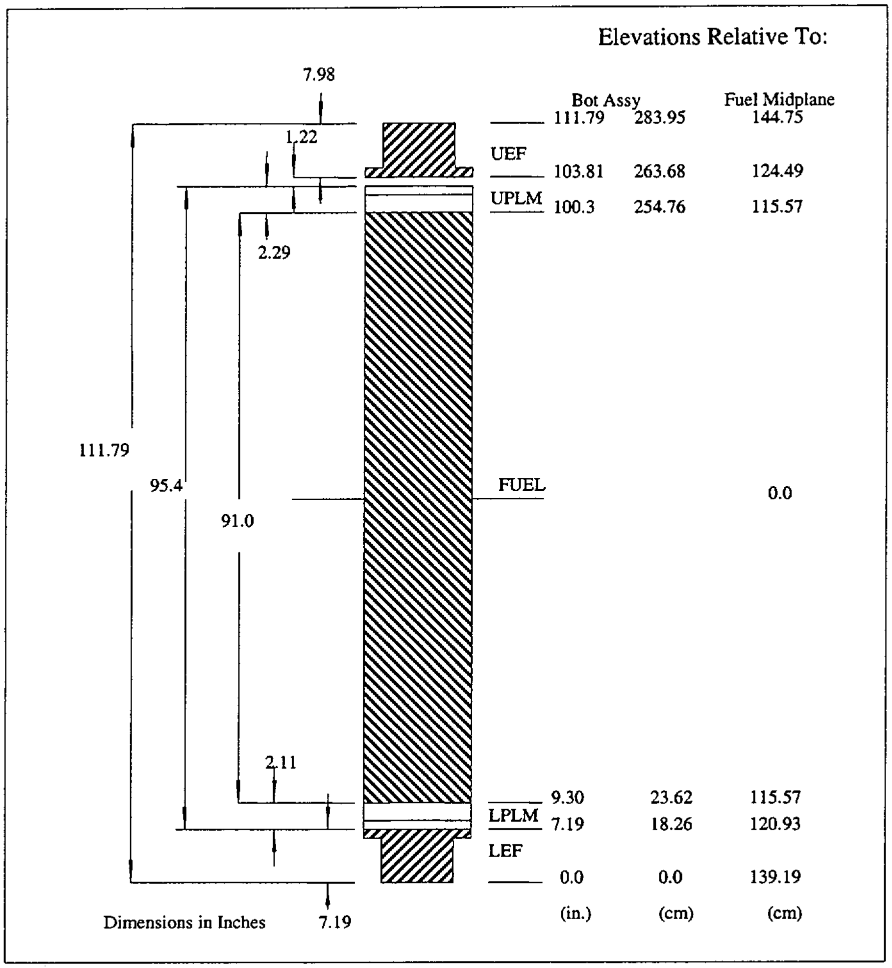
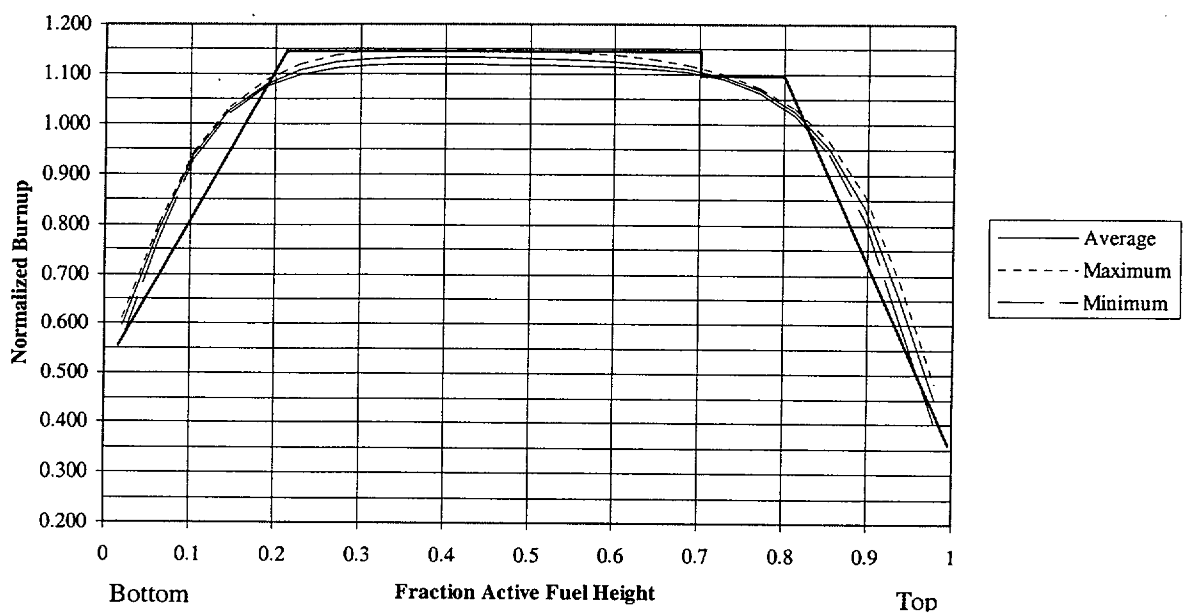


Figure 5.2-2 Yankee Class Combustion Engineering Design Basis Fuel Assembly Source Regions and Elevations



SOURCE	REGION
FUEL	Active fuel
UPLM	Upper Plenum
UEF	Upper End Fitting
LPLM	Lower Plenum
LEF	Lower End Fitting

Figure 5.2-3 NAC-MPC Design Basis Fuel Burnup Profile



Condition: 30,000 - 36,000 MWD/MTU

Figure 5.2-4 NAC-MPC Design Basis Neutron and Gamma Source Axial Profiles

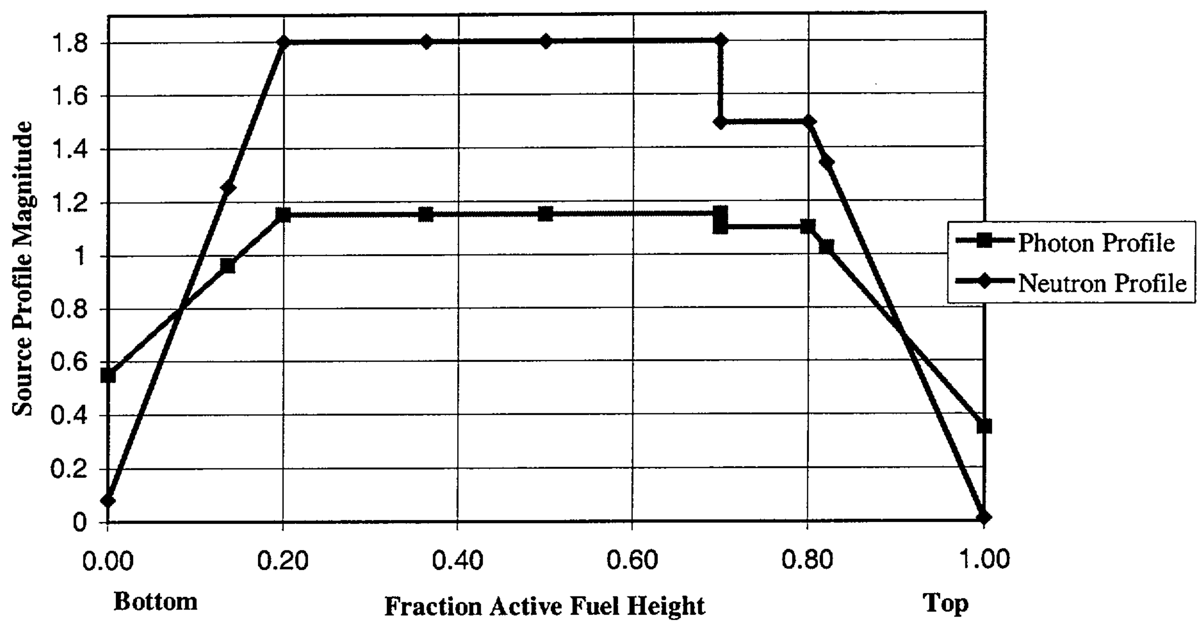


Table 5.2-1 Yankee Class Fuel Assembly Physical Parameters

Parameter	CE Type A	CE Type B	Exxon Type A	Exxon Type B	Exxon Type A	Exxon Type B	Westinghouse Type A	Westinghouse Type B	United Nuclear Type A	United Nuclear Type B
Assembly Configuration	-	-	-	-	-	-	-	-	-	-
Assembly Array	16x16	16x16	16x16	16x16	16x16	16x16	18x18	18x18	16x16	16x16
Max. Enrichment (wt % ^{235}U)	3.90	3.90	4.00	4.00	4.00	4.00	4.94	4.94	4.00	4.00
Min. Enrichment (wt % ^{235}U)	3.50	3.50	3.50	3.50	3.50	3.50	4.94	4.94	4.00	4.00
Max. MTU	0.2394	0.2384	0.2394	0.2384	0.2394	0.2384	0.2869	0.2860	0.2456	0.2446
Fuel Rod Configuration	-	-	-	-	-	-	-	-	-	-
Fuel Rod Pitch (cm)	1.1989	1.1989	1.1989	1.1989	1.1989	1.1989	1.0719	1.0719	1.1887	1.1887
Active Fuel Length (cm)	231.1400	231.1400	231.1400	231.1400	231.1400	231.1400	233.9975	233.9975	231.1400	231.1400
Rod OD (cm)	0.9271	0.9271	0.9271	0.9271	0.9271	0.9271	0.8636	0.8636	0.9271	0.9271
Clad ID (cm)	0.8052	0.8052	0.8052	0.8052	0.8052	0.8052	0.7569	0.7569	0.8052	0.8052
Pellet OD (cm)	0.7887	0.7887	0.7887	0.7887	0.7887	0.7887	0.7468	0.7468	0.7887	0.7887
Diametral Gap (cm)	0.0165	0.0165	0.0165	0.0165	0.0165	0.0165	0.0102	0.0102	0.0165	0.0165
Rods per Assembly	231	230	231	230	231	230	305	304	237	236
Fuel Material	UO_2	UO_2	UO_2	UO_2	UO_2	UO_2	UO_2	UO_2	UO_2	UO_2
Clad Material	Zircaloy	Zircaloy	Zircaloy	Zircaloy	Zircaloy	Zircaloy	SS 348	SS 348	Zircaloy	Zircaloy
Displacement Rod Configuration	-	-	-	-	-	-	-	-	-	-
Displacement Rod Material	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	Zircaloy - 4	Zircaloy - 4
Displacement Rod Diameter (cm)	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	0.9271	0.9271
Number Per Assembly	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	2	2
Guide Bar Configuration	-	-	-	-	-	-	-	-	-	-
Guide Bar Material	Zircaloy - 4	Zircaloy - 4	SS 304L	SS 304L	Zircaloy	Zircaloy	N/A	N/A	N/A	N/A
Guide Bar Width (cm)	1.0973	1.0973	1.0566	1.0566	1.0566	1.0566	N/A	N/A	N/A	N/A
Guide Bar Shape (cm)	Square	Square	Square	Square	Square	Square	N/A	N/A	N/A	N/A
Number Per Assembly	8	8	8	8	8	8	N/A	N/A	N/A	N/A
Instrument Tube Configuration	-	-	-	-	-	-	-	-	-	-
Instrument Tube ID (cm)	0.9970	0.9970	0.9970	0.9970	0.9970	0.9970	0.9995	0.9995	0.9995	0.9995
Instrument Tube OD (cm)	1.1481	1.1481	1.0884	1.0884	1.0884	1.0884	1.0884	1.0884	1.0884	1.0884
Number Per Assembly	1	1	1	1	1	1	1	1	1	1
Instrument Tube Material	Zircaloy - 4	Zircaloy - 4	SS 304	SS 304	Zircaloy	Zircaloy	SS 304	SS 304	SS 304	SS 304

Table 5.2-2 Yankee Class Design Basis Fuel Characteristics for Shielding Evaluations

Parameter	CE Type A
Assembly Configuration	-
Assembly Length (inches)	111.79
Assembly Array	16x16
Minimum Enrichment (wt % ^{235}U)	3.70[3]
UO_2 Fuel Mass (kg) at 95% TD	271.6
Fuel Rod Configuration	-
Fuel Rod Pitch (inches)	0.472
Overall Rod Length (inches)	95.4
Active Fuel Length (inches)	91
Rod OD (inches)	0.365
Clad ID (inches)	0.317
Pellet OD (inches)	0.3105
Diametral Gap (inches)	0.0065
Rods per Assembly	231
Clad Material	Zircaloy
Guide Bar Configuration	-
Guide Bar Material	Zircaloy - 4
Guide Bar Width (inches)	0.432
Guide Bar Length (inches)	96.52
Guide Bar Shape	Square
Number Per Assembly	8
Instrument Tube Configuration	-
Instrument Tube ID (inches)	0.3925
Instrument Tube OD (inches)	0.452
Instrument Tube Length (inches)	97.35
Number Per Assembly	1
Instrument Tube Material	Zircaloy - 4
Hardware Configuration	-
Top Nozzle Material	SS 304
Bottom Nozzle Material	SS 304
Upper Plenum Spring Material	SS 302[1]
Top Nozzle Length (inches)	7.98
Bottom Nozzle Length (inches)	7.19
Upper Plenum Length (inches)	1.942
Top Nozzle Mass (kg)	5.5
Bottom Nozzle Mass (kg)	5.18
Upper Plenum Spring Mass (kg)	0.762
Upper Plenum Grid Mass (kg)	0.590
Lower Plenum Material	Stainless Steel and Inconel[1,2]
Lower Plenum Mass (kg)	1.73[2]
Incore Grid Spacers	Zircaloy -4

Notes: 1. For simplicity, all Inconel and steel are modeled as Type 304 stainless steel.
2. Includes Inconel grid and lower plenum spacer.
3. Combustion Engineering fuel may be loaded at a cool time of 8.0 years with a maximum burnup of 32,000 MWD/MTU and a minimum enrichment of 3.5 wt%.

Table 5.2-3 Yankee Class Design Basis Fuel Reactor Operating Conditions

Assembly Power, MW	8.486
$Temperature_{fuel}$, °K	797
$Temperature_{clad}$, °K	600
$Temperature_{mod}$ °K	551
$Density_{mod}$, g/cc	0.766
Boron, ppm	800
Fuel Burnup, MWD/MTU	36,000
Burnup Cycle, days	2 Cycles of 496.22 days
Down Time, days	60

Table 5.2-4 NAC-MPC Design Basis Fuel Source Terms

Decay Heat, kW	12.5
Active Fuel, photons/sec	6.423+16
Active Fuel, neutrons/sec	2.415+9
Upper End fitting, photons/sec	8.330+13
Upper Plenum, photons/sec	2.309+13
Lower End fitting, photons/sec	7.876+13
Lower Plenum, photons/sec	5.242+13

Condition: 36 Combustion Engineering Yankee Class Fuel Assemblies, 8 Years Cooled, and 36,000 MWD/MTU Burnup.

Table 5.2-5 NAC-MPC Design Basis Fuel Gamma Source Spectra

Group	Gamma Source Spectra[1] Fuel Photons/sec	Gamma Source Spectra[1] Fuel Hardware Photons/sec-kg
1	3.7701E+04	0.0000E+00
2	1.7759E+05	0.0000E+00
3	9.0547E+05	0.0000E+00
4	2.2566E+06	0.0000E+00
5	6.2676E+08	1.0141E-15
6	5.1211E+09	3.3511E+04
7	1.0789E+11	2.1611E+07
8	9.9933E+10	9.5163E-03
9	4.8070E+12	9.1066E+11
10	3.4718E+13	3.2247E+12
11	6.3503E+13	4.3841E+09
12	8.2333E+14	3.8100E+06
13	1.1897E+14	1.0971E+07
14	1.7831E+13	1.7359E+08
15	2.8386E+13	1.3230E+08
16	1.0201E+14	2.6645E+09
17	1.3136E+14	1.1044E+10
18	4.5899E+14	5.5673E+10
Total	1.7842E+15	4.2095E+12

1. 36,000 MWD/MTU and 8 years cool time.

Table 5.2-6 NAC-MPC Design Basis Fuel Neutron Source Spectra

Group	Neutrons/sec[1]
1	1.2290E+06
2	1.4080E+07
3	1.5760E+07
4	8.7930E+06
5	1.1840E+07
6	1.2870E+07
7	2.5190E+06
8	0.0000E+00
9	0.0000E+00
10	0.0000E+00
11	0.0000E+00
12	0.0000E+00
13	0.0000E+00
14	0.0000E+00
15	0.0000E+00
16	0.0000E+00
17	0.0000E+00
18	0.0000E+00
19	0.0000E+00
20	0.0000E+00
21	0.0000E+00
22	0.0000E+00
23	0.0000E+00
24	0.0000E+00
25	0.0000E+00
26	0.0000E+00
27	0.0000E+00
TOTAL	6.7090E+07

1. 36,000 MWD/MTU and 8 year cool time.

5.3 Model Specification

Both one-dimensional SAS1 and three-dimensional SAS4 models are used in the shielding evaluations of the NAC-MPC storage system. The SAS1 radial and axial models are used to estimate the peak and average dose rates on the sides, top and bottom of the storage and transfer casks, and to determine the minimum cool time for the 36,000 MWD/MTU fuel. The one-dimensional models represent the casks as either semi-infinite cylinders or slabs. The method of solution uses the XSDRNPM (Greene) discrete ordinates code and the XSDOSE (Buckholz) flux at a point estimation code. Bucklings are applied to the SAS1 models to account for transverse leakage. One-dimensional analysis also serves as a cross check to the more complex three-dimensional model results.

The SAS4 three-dimensional shielding models are used to estimate the dose profiles at the surfaces of the cask and at streaming paths such as the storage cask inlets and outlets, or the canister vent and drain ports. The method of solution is adjoint discrete ordinates and Monte Carlo (Tang) using the XSDRNPM and MORSE codes, respectively. Since SAS4 requires model symmetry at the fuel midplane, two models are created for each cask, a top and a bottom model. Radial biasing is performed to estimate dose rates on the sides of the cask, and axial biasing is performed to estimate dose rates on the top and bottom surfaces of the cask. Modifications are made to SAS4 to tally dose rates all along the radial, top and bottom surfaces of the cask as well as any cylindrical surface surrounding the cask. Thus, detailed dose rate profiles are determined that explicitly show peaks due to the fuel burnup profile, activated hardware gamma emission and streaming paths.

In both SAS1 and SAS4 models, the fuel and hardware source regions are homogenized within the volumes defined by the periphery of the basket tubes and by the elevations of the basket heat transfer zone, the active fuel, the plenum and the end fittings (Table 5.3-1). Within these volumes, the material masses of the fuel assembly and basket are homogenized (Table 5.3-2). The design basis fuel assembly and the NAC-MPC basket materials are obtained from Table 5.2-2 and from drawings 455-881, 455-891, 455-892, 455-893, 455-894 and 455-895 (Section 1.5). The SCALE 4.3 standard composition library (Landers) default compositions and isotopic distributions are used except for BORAL, NS-4-FR and concrete (Section 5.3.2). The resultant material zones and nuclide densities are summarized in Table 5.3-3. In all models, the cask and canister shield thicknesses are explicitly represented.

Both the SAS1 and SAS4 models utilize fuel midplane symmetry. Thus, all shielding models are developed with respect to the fuel midplane as the origin. This symmetry is required in the SAS4 models due to the automated biasing techniques employed and because the dose rate tallies from the symmetric halves are averaged together for computational efficiency.

5.3.1 Description of Radial and Axial Shielding Configurations

The NAC-MPC storage cask has an interior cavity with a radius of 39.5 inches (100.33 cm). Radial shielding consists of a 3.5-inch (8.89 cm) carbon steel shell surrounded by 21 inches (53.34 cm) of concrete. Gamma shielding is provided by both the carbon steel and concrete, and neutron shielding is provided primarily by the concrete. An additional 0.625 inch (1.59 cm) of stainless steel is provided by the canister shell for radial gamma shielding. The storage cask top shielding comprises 8 inches (20.32 cm) of stainless steel from the canister lids, 4.125 inches (10.48 cm) of carbon steel from the shield plug which encloses 1 inch (2.54 cm) of NS-4-FR and finally, 1.5 inches (3.81 cm) of carbon steel from the storage cask lid. The bottom of the storage cask rests on the concrete pad and is inaccessible. In the case of the storage cask inlets, some shielding is provided by the storage cask structural components. These are 2 inches (5.08 cm) of carbon steel from the pedestal plate, 1 inch (2.54 cm) of carbon steel from the cask base plate and 1 inch (2.54 cm) of stainless steel from the canister bottom plate.

The NAC-MPC transfer cask has an inside radius of 35.75 inches (90.81 cm) and has a multiwall radial shield design consisting of 0.75 inch (1.91 cm) of carbon steel, 3.5 inches (8.89 cm) of lead, 2 inches (5.08 cm) of a solid borated polymer (NS-4-FR), and 1.25 inches (3.18 cm) of carbon steel. Gamma shielding is provided by the steel and lead layers, and neutron shielding is provided primarily by the NS-4-FR. An additional 0.625 inch (1.59 cm) of stainless steel gamma shielding is provided by the canister shell. The transfer cask bottom shield design comprises carbon steel doors 9.50 inches (24.13 cm) thick. The top shielding of the transfer cask is provided by the 5-inch (12.70 cm) stainless steel shield lid and the 3-inch (7.62 cm) stainless steel structural lid. In addition, a 5-inch (12.70 cm) carbon steel temporary shield is used during welding, draining, drying and transfer operations. This temporary shielding is removed prior to storage.

5.3.1.1 One-Dimensional Radial and Axial Shielding Models

Since the fuel assembly and basket features are not explicitly modeled in one-dimensional analysis, the fuel/basket interior is modeled as a set of homogenized material volumes based on equivalent cylindrical volumes. These volumes are defined by the areas created by: the central basket hole; the periphery of the basket tubes and the edge of the steel support disks; and by the elevations created by the basket heat transfer zone, the active fuel, the fuel assembly plenums and the fuel assembly end fittings.

The NAC-MPC fuel basket is divided into three radial regions: a central hole (void), fuel/basket region, and a basket/disk region. These regions have equivalent radii of 4.66, 30.63 and 34.51 inches (11.83, 77.80, and 87.66 cm), respectively. Axially, the basket is divided into seven regions: the top, middle and bottom fuel/basket regions; the upper and lower plenum/basket regions; and the upper and lower end fittings/basket regions. For the top models, the top fuel, top plenum and top end fitting regions have elevations with respect to the fuel midplane of 45.50, 49.01, and 56.99 inches (115.57, 124.49 and 144.75 cm), respectively. For the bottom models, the bottom fuel, bottom plenum and bottom end fitting regions have elevations with respect to the fuel midplane of 45.50, 47.61, and 54.80 inches (115.57, 120.93 and 139.19 cm), respectively.

In each of these regions, the relevant fuel assembly material and any basket material present are homogenized. Basket materials include the steel support disks, aluminum heat transfer disks, top and bottom weldments, fuel tubes, BORAL sheets, and BORAL cover sheets. Fuel assembly materials include: UO_2, cladding, grids, plenum springs and spacers, and end-fittings. The resultant material and nuclide densities are described in Section 5.3.2.

The one-dimensional radial models of the storage cask and the transfer cask are based on the cylindrical representation of the fuel/basket source regions (previously described) surrounded by the explicit canister and cask radial shield dimensions. An axial buckling equal to the active fuel height of 91 inches (231.14 cm) is assumed for all radial models.

The one-dimensional top and bottom axial models of the storage and transfer casks are based on a slab representation of the fuel/basket axial regions covered by the explicitly modeled canister and storage cask axial shield regions. As previously stated, the one-dimensional axial model elevations are specified from the active fuel centerline, which SAS1 automatically establishes as the reflecting boundary. Two models are utilized for each cask: one from the active fuel centerline to the top of the cask; and one from the active fuel centerline to the base plate of the cask. Two transverse bucklings equal to the fuel/basket zone equivalent diameter of 61.26 inches (155.6 cm) are assumed for both axial models.

5.3.1.2 Three-Dimensional Top and Bottom Shielding Models

SAS4 three-dimensional shielding analysis allows detailed modeling of the fuel assemblies, basket and cask shield configuration including streaming paths. Some fuel assembly and basket detail is homogenized to simplify model input and improve computational efficiency. Thus, the three-dimensional models maintain the equivalent fuel/basket source volumes developed for the one-dimensional models, but explicitly model the radial and axial extent of the source regions and the cask body details. As in the SAS1 models, the fuel and hardware source regions are homogenized within the volumes defined by the periphery of the basket tubes and by the elevations of the basket heat transfer zone, the active fuel, the plenum and the end fittings. Cask body details include the true axial extent of the cask shields as described by the drawings in Section 1.5, as well as radiation streaming paths such as the storage cask inlets and outlets and the canister vent and drain ports.

SAS4 requires cask model symmetry at the fuel midplane due to the nature of the automated biasing techniques employed and because dose rate tallies from the symmetric halves of the model are averaged together for computational efficiency. Thus, two models are created for each cask, a top and a bottom model. As in the SAS1 models, all three-dimensional shielding models are developed with respect to the fuel midplane as the origin.

The geometry of SAS4 is based on MARS (West) combinatorial geometry embedded in the MORSE code (Emmett). In this geometry, bodies such as cylinders and rectangular parallelepipeds are used to describe the extent of zones of material. Zones are volumes of constant material (cross sections) and are defined by logical operations on geometric bodies.

SAS4 employs an automated biasing technique for the MORSE Monte Carlo calculations based on either a radial or an axial XSDRNPM adjoint calculation. In the case of radial biasing, the adjoint calculation is performed for the radial shields and corresponding fuel/basket regions. In the case of axial biasing, the adjoint calculation is performed for the top or bottom shields and corresponding axial fuel/basket regions. Radial biasing is employed to improve the Monte Carlo computational efficiency and dose rate statistics on the sides of the cask. Axial biasing is employed to improve Monte Carlo computational efficiency and dose rate statistics on the top or bottom surfaces of the cask. The dose rate profiles resulting from both radial and axial biasing calculations yield a complete dose profile of the entire cask with design basis fuel.

MORSE Monte Carlo calculations are performed for each type of source in each source region. In the case of the NAC-MPC basket and design basis fuel assembly configuration, this leads to eight source terms: middle fuel neutron, gamma and n-gamma; top or bottom fuel neutron, gamma and n-gamma; activated plenum hardware gamma and activated end fitting gamma. Twenty to thirty million histories (gamma and neutron combined) are tracked to yield a dose rate surface profile for each surface.

5.3.1.2.1 NAC-MPC Storage Cask Three-Dimensional Models

The three-dimensional top model of the NAC-MPC storage cask containing 36 design basis Yankee Class fuel assemblies is based on the homogenized cylindrical representation of the basket, and the following top features of the storage cask:

- Heat transfer annulus
- Carbon steel shell with four cutouts for outlet vents
- Concrete shield with four cutouts for outlet vents
- Four outlet vents including carbon steel lining
- Carbon steel shield plug
- Shield plug neutron shield
- Carbon steel top lid

Details on the elevations and radii used in creating the three-dimensional top model are taken directly from the drawings in Section 1.5. Elevations associated with the storage cask three-dimensional features are established with respect to the active fuel midplane of the Yankee Class fuel assembly (Figure 5.2-2) for the combinatorial model. The three-dimensional storage cask top model is shown in Figure 5.3-1. The MARS geometry requires 71 bodies (23 right circular cylinders and 48 rectangular parallelepipeds) to define 39 model zones with combinatorial logic.

The three-dimensional bottom model of the NAC-MPC storage cask is based on the homogenized cylindrical representation of the fuel/basket and the following bottom features of the storage cask:

- Heat transfer annulus
- Carbon steel shell with four cutouts for inlet vents
- Concrete shield with four cutouts for inlet vents
- Four inlet vents with carbon steel linings
- Carbon steel bottom base plate
- Carbon steel support stand with four cutouts for air flow
- Carbon steel shield ring
- Carbon steel storage cask bottom
- Concrete pad below base plate

The three-dimensional storage cask bottom model is shown in Figure 5.3-2. The MARS geometry requires 55 bodies (30 right circular cylinders and 25 rectangular parallelepipeds) to define 50 model zones with combinatorial logic.

5.3.1.2.2 NAC-MPC Transfer Cask Three-Dimensional Models

Several different three-dimensional models of the top portion of the transfer cask are used in the shielding evaluations. These include wet and dry cavity conditions as well as the corresponding shield lid and structural lid placement. The top configuration of the transfer cask is evaluated in detail for the welding, draining and drying operations. As with the storage cask models, top models of the NAC-MPC transfer cask containing 36 design basis Yankee fuel assemblies are based on a homogenized representation of the basket, but the rectangular periphery of the basket source region is modeled in order to more accurately estimate vent and drain port dose rates.

The basket disks outside the fuel/basket region are explicitly modeled to more accurately account for basket streaming. The following features of the transfer cask are considered:

- Vent and drain port openings in the shield lid
- Upper weldment shield ring
- Edge tapering and port cutouts in the temporary shielding
- Two lifting trunnions cut through the radial shield to the inner shell
- Lead and neutron shielding overlap at the top as shown on the transfer cask drawings

Details of the elevations and radii used in creating the three-dimensional top model are taken directly from the drawings in Section 1.5. As with the other three-dimensional models, elevations associated with the transfer cask three-dimensional features are established with respect to the active fuel midplane of the Yankee Class fuel assembly for the combinatorial geometry model. The three-dimensional transfer cask top model including shield and structural lid installation is shown in Figure 5.3-3. The MARS geometry requires 108 bodies (94 right circular cylinders and 14 rectangular parallelepipeds) to define 68 model zones.

The three-dimensional bottom model of the NAC-MPC transfer cask is based on the same homogenized representation of the fuel/basket as the top model. As with the top model of the transfer cask, the evaluations include both wet and dry canister cavity. The following bottom features of the transfer cask are considered:

- Termination of the radial shields at the bottom doors
- 9.5 inches of carbon steel shielding representing the bottom doors

The transfer cask bottom model is shown in Figure 5.3-4. The MARS geometry requires 69 bodies (56 right circular cylinders and 13 rectangular parallelepipeds) to define 59 model zones with combinatorial logic.

5.3.2 Shield Regional Densities

The SCALE 4.3 standard composition library (Landers) default compositions and isotopic distributions are used unless otherwise indicated. The composition densities before homogenization are:

Material	Density (g/cc)
UO_2 at 95% TD	- 10.412
Zircaloy	- 6.56
H_2O	- 0.9982
Stainless Steel 304	- 7.92
Lead	- 11.344
Aluminum	- 2.702
BORAL (core)	- 2.623 (non-standard)
NS-4-FR	- 1.629 (non-standard)
Concrete	- 2.243 (based on 140 lb/ft^3 design spec. minimum)
Carbon Steel	- 7.821

Reinforcing steel is conservatively ignored in the concrete density. Basket and fuel assembly regional volumes are shown in Table 5.3-1 and the regional masses are shown in Table 5.3-2. The resultant regional homogenized densities for the design basis CE fuel assembly and shield densities are provided in Table 5.3-3.

"Figure5.3-1 NAC-MPC Storage Cask Three-Dimensional Top Model

FIGURE WITHHELD UNDER 10 CFR 2.390

Figure 5.3-2 NAC-MPC Storage Cask Three-Dimensional Bottom Model

FIGURE WITHHELD UNDER 10 CFR 2.390

Figure 5.3-3 NAC-MPC Transfer Cask Three-Dimensional Top Model Including Shield and Structural Lid

FIGURE WITHHELD UNDER 10 CFR 2.390

Figure 5.3-4 NAC-MPC Transfer Cask Three-Dimensional Bottom Model

FIGURE WITHHELD UNDER 10 CFR 2.390

Table 5.3-1 Fuel/Basket Regional Volumes (cm^3)

Axial Zone[1]	Fuel/Basket Region	Basket/Disk Region
LEF	3.3927E+05	9.3029E+04
LPLM	9.9562E+04	2.7301E+04
$FUEL_{bot}$	5.8652E+05	1.6083E+05
$FUEL_{mid}$	2.9368E+06	8.0530E+05
$FUEL_{top}$	7.7054E+05	2.1129E+05
UPLM	1.6562E+05	4.5415E+04
UEF	3.7654E+05	1.0325E+05

1. See Figure 5.2-2 for zone locations.

Table 5.3-2 Fuel/Basket Regional Masses (kg)

	Fuel/Basket Region						Basket/Disk Region	
	Fuel Assembly Contribution			Basket Contribution				
Axial Zone[1]	UO_2	Zirc	SS 304	SS 304	Al	B_4C	SS 304	Al
LEF	--	0.00	186.48	141.37	--	--	51.24	--
LPLM	--	61.01	62.28	42.55	--	--	51.24	--
$FUEL_{bot}$	1335.47	359.43	0.00	173.01	36.46	5.90	102.47	--
$FUEL_{mid}$	6687.02	1884.71	0.00	1004.41	320.40	29.55	768.52	235.15
$FUEL_{top}$	1754.48	472.20	0.00	237.61	47.89	7.75	153.70	--
UPLM	--	94.69	27.43	52.41	--	--	51.24	--
UEF	--	0.0000	198.00	56.15	--	--	51.24	--

1. See Figure 5.2-2 for zone locations.

Table 5.3-3 Regional Homogenized Densities and Shield Densities

Zone/Material	Density (g/cc)	Nuclides	Density (atom/b-cm)
Middle Fuel Zone			
UO_2	2.2769	^{234}U	2.793E-07
		^{235}U	3.656E-05
		^{238}U	5.041E-03
		O	1.016E-02
Zircaloy	0.6417	Zr	4.236E-03
SS 304	0.3420	Cr	7.526E-04
		Mn	7.498E-05
		Fe	2.563E-03
		Ni	3.334E-04
Aluminum	0.1091	Al	2.435E-03
B_4C	0.0101	^{10}B	8.764E-05
		^{11}B	3.528E-04
		C	1.101E-04
H_2O Wet Transfer Cask	0.6000	H	4.010E-02
		O	3.021E-02
Middle Basket/Disk Zone			
SS 304	0.9543	Cr	2.100E-03
		Mn	2.092E-04
		Fe	7.152E-03
		Ni	9.303E-04
Aluminum	0.2920	Al	6.517E-03
H_2O Wet Transfer Cask	0.7700	H	5.151E-02
		O	2.575E-02
Steel Shielding			
SS 304	7.9200	Cr	1.743E-02
		Mn	1.736E-03
		Fe	5.9336-02
		Ni	7.721E-03
Carbon Steel Shielding			
Carbon Steel	7.8212	Fe	8.350E-02
		C	3.925E-03
Concrete Shielding			
Concrete	2.2430	Fe	3.386E-04
		H	1.340E-02
		Al	1.702E-03
		Ca	1.483E-03
		O	4.494E-02
		Na	1.704E-03
		N	1.621E-02

Table 5.3-3 Regional Homogenized Densities and Shield Densities (Continued)

Material/Zone	Density (g/cc)	Nuclides	Density (atom/b-cm)
Top Fuel/Basket Zone			
UO_2	2.2769	^{234}U	2.793E-07
		^{235}U	3.656E-05
		^{238}U	5.041E-03
		O	1.016E-02
Zircaloy	0.6128	Zr	4.046E-03
SS 304	0.3084	Cr	6.787E-04
		Mn	6.761E-05
		Fe	2.311E-03
		Ni	3.006E-04
Aluminum	0.0622	Al	1.388E-03
B_4C	0.0101	^{10}B	8.764E-05
		^{11}B	3.528E-04
		C	1.101E-04
H_2O Transfer Cask Wet	0.6303	H	4.214E-02
		O	3.123E-02
Top Plenum Zone			
Zircaloy	0.5718	Zr	3.775E-03
SS 304	0.4821	Cr	1.485E-03
		Mn	1.057E-04
		Fe	3.613E-03
		Ni	4.700E-04
H_2O Transfer Cask Wet	0.7019	H	4.695E-02
		O	2.348E-02
Top End Fitting Zone			
SS 304	0.6749	Cr	1.743E-02
		Mn	1.480E-04
		Fe	5.058E-03
		Ni	6.579E-04
H_2O Transfer Cask Wet	0.9132	H	6.108E-02
		O	3.054E-02
Neutron Shield			
NS-4-FR	1.6291	^{10}B	8.553E-05
		^{11}B	3.422E-04
		Al	7.763E-03
		H	5.854E-02
		O	2.609E-02
		N	1.394E-03
		C	2.264E-02

Table 5.3-3 Regional Homogenized Densities and Shield Densities (Continued)

Material/Zone	Density (g/cc)	Nuclides	Density (atom/b-cm)
Bottom Fuel/Basket Zone			
UO_2	2.2769	^{234}U	2.793E-07
		^{235}U	3.656E-05
		^{238}U	5.041E-03
		O	1.016E-02
Zircaloy	0.6128	Zr	4.046E-03
SS 304	0.2350	Cr	6.492E-04
		Mn	6.467E-05
		Fe	2.211E-03
		Ni	2.876E-04
Aluminum	0.0622	Al	1.388E-03
B_4C	0.0101	^{10}B	8.764E-05
		^{11}B	3.528E-04
		C	1.101E-04
H_2O Transfer Cask Wet	0.6322	H	4.228E-02
		O	3.130E-02
Bottom Plenum Zone			
Zircaloy	0.6128	Zr	4.046E-03
SS 304	1.0529	Cr	2.317E-03
		Mn	2.308E-03
		Fe	7.891E-03
		Ni	1.026E-03
H_2O Transfer Cask Wet	0.5447	H	3.644E-02
		O	1.822E-02
Bottom End fitting Zone			
SS 304	0.9664	Cr	2.127E-03
		Mn	2.119E-04
		Fe	7.243E-03
		Ni	9.421E-04
H_2O Transfer Cask Wet	0.8764	H	5.862E-02
		O	2.931E-02

5.4 Shielding Evaluation

This section evaluates the shielding design of NAC-MPC transfer and storage casks. The calculational methods are described. Shielding calculations are performed with design basis Yankee Class fuel source terms at 36,000 MWD/MTU and 8 years cooling time. Dose rate profiles are reported as a function of distance from the sides, top, and bottom of the NAC-MPC storage and transfer casks. Storage cask shielded source terms (neutron and gamma fluxes at the cask surface) are provided for ISFSI controlled area boundary dose evaluations.

5.4.1 Calculational Methods

Shielding evaluations of the transfer and storage casks are performed with SCALE 4.3 for the PC. In particular, SCALE shielding analysis sequence SAS2H is used to generate source terms for the design basis fuel. SAS1 is used to perform one-dimensional radial and axial shielding analysis, and a modified version of SAS4 is used to perform three-dimensional shielding analysis. The coupled 27 group neutron, 18 group gamma ENDF/B-IV (27N-18COUPLE) cross section library is used in all shielding evaluations. Source terms include fuel neutron, fuel gamma, and gamma contributions from activated hardware. Dose rate evaluations include the effect of fuel burnup peaking on fuel neutron and gamma source terms. The SCALE shielding analysis sequences and cross section libraries recently have been benchmarked to measurements of light water reactor fuel source terms, shielding material dose rate attenuation, and spent fuel storage and transport cask dose rates (Broadhead).

The SAS2H code sequence is used to generate source terms for the Yankee Class design basis fuel. SAS2H includes an XSDRNPM neutronics model of the fuel assembly and ORIGEN-S fuel depletion/source terms calculations. The 27 energy group ENDF/B-IV neutron cross section library, 27GROUPNDF4, is used in the source terms calculations. Source terms are generated for both UO_2 fuel and fuel assembly hardware. The hardware activation is calculated by ORIGEN-S using the incore neutron flux spectrum produced by the SAS2H neutronics model. The hardware is assumed to be Type 304 stainless steel with 1.2 g/kg ^{59}Co impurity. The effects of axial flux spectrum and magnitude variation on hardware activation are estimated by flux ratios based on empirical data (Luskic).

Both the one-dimensional SAS1 and the three-dimensional SAS4 shielding models are used in the evaluations of the NAC-MPC storage system. The SAS1 radial and axial models are used to estimate the peak and average dose rates on the sides, top, and bottom of the storage and transfer casks. The models represent the cask as either semi-infinite cylinders or slabs. The method of solution is XSDRNPM discrete ordinates. Bucklings are applied to the one-dimensional models to account for transverse leakage. One-dimensional analysis also serves as a cross check of the three-dimensional model results and is employed to establish the minimum cool time for the loading of 32,000 MWD/MTU burnup assemblies and the Exxon 36,000 MWD/MTU assemblies.

The SAS4 shielding models are used to estimate the dose profiles along the surfaces of the transfer and storage casks and to estimate doses in and around streaming paths such as the storage cask inlets and outlets, and the canister vent and drain ports. The SAS4 models represent the cask body and any streaming paths with combinatorial logic. The method of solution is adjoint discrete ordinates and Monte Carlo using the XSDRNPM and MORSE codes, respectively. Since SAS4 requires model symmetry at the fuel midplane, two models are created for each cask, a top and a bottom model. Radial biasing is performed to estimate dose rate on the sides of the cask, and axial biasing is performed to estimate dose rates on the top and bottom surfaces of the cask. Modifications are made to SAS4 to determine dose rates all along the radial, top and bottom surfaces of the cask as well as any cylindrical surface surrounding the cask. Thus, detailed dose profiles are determined that explicitly show peaks due to the fuel burnup profile, activated hardware gamma emission and any streaming paths.

In both the SAS1 and SAS4 models, the fuel and hardware source regions are homogenized within the volumes described by the periphery of the basket tubes, and defined by the fuel assembly active fuel, plenum, and end fitting elevations. Within these volumes, the material masses of the fuel assembly and basket are preserved.

5.4.2 Flux-to-Dose Rate Conversion Factors

The ANSI/ANS 6.1.1-1977 flux-to-dose rate conversion factors are used in all NAC-MPC shielding evaluations. These factors are default for SCALE 4.3. Tables 5.4-1 and 5.4-2 show the group flux-to-dose rate factors associated with the coupled 27 group neutron and 18 group gamma cross section library used in the shielding evaluations.

5.4.3 Dose Rates

This section provides detailed dose rate profiles for the NAC-MPC storage and transfer cask based on the source terms presented in Section 5.2. Design basis fuel source terms include contributions from fuel neutron, fuel gamma and activated hardware gamma. The fuel assembly activated hardware gamma source terms include: steel and inconel in the upper and lower fuel assembly end fittings, and upper and lower fuel rod plenum hardware. Peaking factors of 1.15 and 1.80 are applied to the one-dimensional radial fuel gamma and neutron dose rates, respectively. The three-dimensional model dose rates include the axial profiles for neutron and gamma source distributions shown in Figure 5.2-4.

5.4.3.1 One-Dimensional Storage Cask Dose Rates

One-dimensional radial dose rates with design basis fuel were found to be in good agreement with the three-dimensional models at the radial midplane. However, the peaks in the radial dose rates, due to activated endfittings, cannot be captured by one-dimensional analysis. One-dimensional dose rates at the top of the storage cask are significantly lower than those calculated using three-dimensional analysis. This is primarily due to the neutron component of the dose rates and the transverse bucklings applied in the one-dimensional axial model as well as streaming effects caused by the heat transfer annulus and top vents. Except for the neutron component of the top axial model and obvious limitations in geometry, one-dimensional analysis is found to support the results of the more complicated three-dimensional models. Except for the storage cask loss of concrete accident radial dose rates shown in Table 5.4-3, the dose rate results from three-dimensional analysis are reported.

One-dimensional radial surface dose rates are also used to determine the minimum cool times, based on the design basis fuel values, for CE, UN, and WE fuel types with a maximum burnup of 32,000 MWD/MTU and Exxon assemblies at 36,000 MWD/MTU. The calculated minimum cool times for CE, UN, and WE fuel at 32,000 MWD/MTU are 8, 13 and 21 years, respectively. Exxon assemblies with steel hardware require 16 years cooling while the Exxon assemblies with Zircaloy hardware require only 9 years cooling.

5.4.3.2 Three-Dimensional Storage Cask Dose Rates

The NAC-MPC storage cask three-dimensional model dose rates are presented in Figures 5.4-1 through 5.4-7. Approximately 50 million particle histories (neutron and gamma) are tracked to yield the dose rate profiles presented in theses figures. The average standard deviation for the side total dose rate shown in Figures 5.4-1 and 5.4-2 is less than ± 2%, and the average standard deviation for the top total dose rate shown in Figures 5.4-6 and 5.4-7 is less than ± 5%. The average standard deviation for the inlet and outlet vent total dose rates shown in Figures 5.4-3, 5.4-4 and 5.4-5 is less than ± 5%, and the standard deviation for the peak dose rates at the vent opening are less than ± 10%.

The vertical profile along the radial surface of the storage cask, as well as at distances of 30.48 cm (1 foot), 1 meter, and 2 meters from it, are plotted in Figure 5.4-1 as a function of elevation. Each datum represents the circumferentially average dose rate at the corresponding distance and elevations. The negative elevations are the dose rates from the bottom model computations, while the positive elevations are the dose rates from the top model computations. The discontinuity observed at zero elevation (midplane of the fuel) is a modeling artifact due to the decoupling of the upper and lower portions of the cask. 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 intake and upper outlet vents. The average and maximum side surface dose rate for the storage cask are 37 (0.3%) and 47.3 (0.4%) mrem/hr, respectively.

The radial surface dose profile is further described by source component in Figure 5.4-2. The source components in both models contribute to the radial doses largely as one would expect, i.e., at the elevations where they are located. Since these doses are circumferential averages, the detailed circumferential dose rate profile at the top vent elevation and the bottom vent inlet are shown radially in Figure 5.4-3 and Figure 5.4-4, respectively. The dose rates shown in Figures 5.4-3 and 5.4-4 were computed using a variance-weighted average of the dose rates in the four symmetric quadrants at the vent elevation. A maximum dose rate of 24 mrem/hr (5%) is calculated at the surface of the outlet vent and a maximum dose rate of 99 (5.4%) mrem/hr was calculated at the entrance of the inlet vent.

In Figure 5.4-5, the circumferential dose rate profile at the support ring cutout elevation is shown on the storage surface and at distances of 30.48 cm (1 foot), 1 meter, and 2 meters from the surface. The peak in the circumferential dose rate is not at the location of the cutout, but above the inlet vent location. Note that these peak dose rates are higher at 1 foot from the storage cask than they are on the surface of the storage cask. This is due to photon scattering off the storage cask concrete base through the inlet vent opening and up to the cutout elevation.

The dose rate profiles on the top surface of the storage cask and at distances of 1 foot and 1 meter above the lid are shown in Figure 5.4-6. The dose rates are plotted radially out from the centerline of the storage cask. Two dose rate peaks are observed on the storage cask top surface: one in the vicinity of 90 to 100 cm, which corresponds to the location of the heat transfer annular gap and another at approximately 130 cm. The dose rate profile on the top surface of the storage cask is shown by source component in Figure 5.4-7 and indicates that the peak dose rate above the annular gap is caused by neutrons streaming up the gap. The component profile also indicates that the second peak is created by gammas from the end fitting, top fuel, and top plenum source regions. This peak occurs at approximately the same radial location as the vertical leg in the upper outlet vent. Thus, it is a result of a decrease in effective shield thickness caused by the void in the concrete due to the outlet vents. The average dose rate over the top of the storage cask is computed to be 25.1 mrem/hr (1.2%), while the peak dose on top of the storage cask is 54 mrem/hr (4.9%).

5.4.3.3 One-Dimensional Transfer Cask Dose Rates

One-dimensional radial dose rates with design basis fuel are in good agreement with the three-dimensional models at the radial midplane. As with the storage cask one-dimensional radial model, the peaks in the radial dose rates due to activated endfittings cannot be captured by one-dimensional analysis. One-dimensional top dose rates at the top and bottom of the transfer cask were significantly lower than three-dimensional analysis. This was primarily due to the neutron component of the dose rates and the transverse bucklings applied in the one-dimensional axial model as well as streaming effects around the temporary shielding. Except for the neutron component of the top axial model, one-dimensional analysis supports the results of the more complicated three-dimensional models.

5.4.3.4 Three-Dimensional Transfer Cask Dose Rates

The transfer cask three-dimensional model dose rates are presented in Figures 5.4-8 through 5.4-15. Approximately 100 million particle histories (neutron and gamma) are tracked to yield the dose rate profiles presented in theses figures. The average standard deviation for the side total dose rates shown in Figures 5.4-8 through 5.4-11 is less than ± 2%, and the average standard deviation for the top total dose rates shown in Figures 5.4-12 through 5.4-15 is less than ± 2%.

The transfer cask side dose rate profiles with a wet cavity are shown in Figure 5.4-8 as a function of distance and in Figure 5.4-9 as a function of source component. In this condition, the majority of the dose rate is from fuel gamma and activated end fitting gamma. It is assumed in the model that the water level in the canister is lowered for welding operations, thus, the top end fitting is uncovered and causes a large peak in dose rate at the top of the transfer cask due to the gamma source from the activated top end fitting. In this condition, the peak and average dose rates on the side of the transfer cask are 210.2 (0.8%) and 79.5 (0.3%) mrem/hr, respectively, and the peak and average dose rates at 1 meter are 40.5 (0.7%) and 26.4 (0.2 %) mrem/hr, respectively.

The transfer cask side dose rate profiles with a dry cavity are shown in Figure 5.4-10 as a function of distance and in Figure 5.4-11 as a function of source component. In this condition, the majority of the dose rate is from fuel neutron and gamma source, but significant peaks are shown from the activated end fittings. The peak and average dose rates on the side of the transfer cask are 413.4 (1.5%) and 226.3 (0.2%) mrem/hr, respectively, and the peak and average dose rates at 1 meter are 103.4 (0.6%) and 72.2 (0.2 %) mrem/hr, respectively.

The transfer cask peak and average dose rate on the temporary shield surface are 188.7 (1.1%) and 172.0 (0.3%) mrem/hr, respectively. In this condition, the majority of the dose rate is from the activated top end fitting. The peak and average dose rate at 1 meter are 389.1 (5.1%) and 263.7 (1.5%) mrem/hr, respectively.

In the final configuration, the canister cavity is dry, the shield lid and structural lid are in place, and 5" of temporary steel shielding is installed. In this condition, the transfer cask top dose rate are shown in Figure 5.4-12 as a function of distance and in Figure 5.4-13 as a function of component. The majority of the dose rate is from the fuel neutron. The dose rate peaks at the lid edge due to gamma streaming around the tapered edge of the temporary shield. The peak and average dose rates on the top of the transfer cask are 358.9 (2.6%) and 224.6 (0.9%) mrem/hr, respectively, and the peak and average dose rates at 1 meter are 39.6 (4.5%) and 34.2 (1.3 %) mrem/hr, respectively.

The transfer cask bottom dose rate profiles with the cavity wet and dry are shown in Figures 5.4-14 and 5.4-15, respectively. In the wet cavity situation, the peak and average dose rates on the bottom of the transfer cask are 77.2 (0.7%) and 55.9 (0.2 %) mrem/hr, respectively, and the peak and average dose rates at 1 meter are 19.0 (2.1%) and 12.2 (0.3 %) mrem/hr, respectively. In the dry cavity situation, the peak and average dose rates on the bottom of the transfer cask are 398.0 (3.9%) and 194.7 (0.3 %) mrem/hr, respectively, and the peak and average dose rates at 1 meter are 66.7 (3.6%) and 28.2 (0.4 %) mrem/hr, respectively.

5.4.4 Storage Cask Shielded Source Terms

The storage cask shielded source terms are provided in this section for use in the ISFSI controlled area boundary dose evaluations. These shielded source terms are the neutron and gamma fluxes at the surface of the NAC-MPC storage cask due to the neutron and gamma sources specified in Section 5.2. The cask surface fluxes are obtained from one-dimensional SAS1 radial and axial shielding evaluations. These fluxes are in the 27 group neutron and 18 group gamma energy group structure consistent with the SCALE 4.3 27N-28COUPLE cross section library. The group wise fluxes are listed in Tables 5.4-11 through 5.4-13 for the side and the top of the storage cask. At the bottom of each column is the total source strength for use in SKYSHINE-III. This source strength, in the case of the radial component, is based on the surface area of the side storage cask at the active fuel region, and, in the case of the top axial component, is based on the surface area of the top lid. The total source strengths and spectra are used in the SKYSHINE-III direct and air-scatter dose evaluations presented in Chapter 10 for an array of NAC-MPC storage casks at an ISFSI.

Figure 5.4-1 Storage Cask Radial Dose Rate Profile

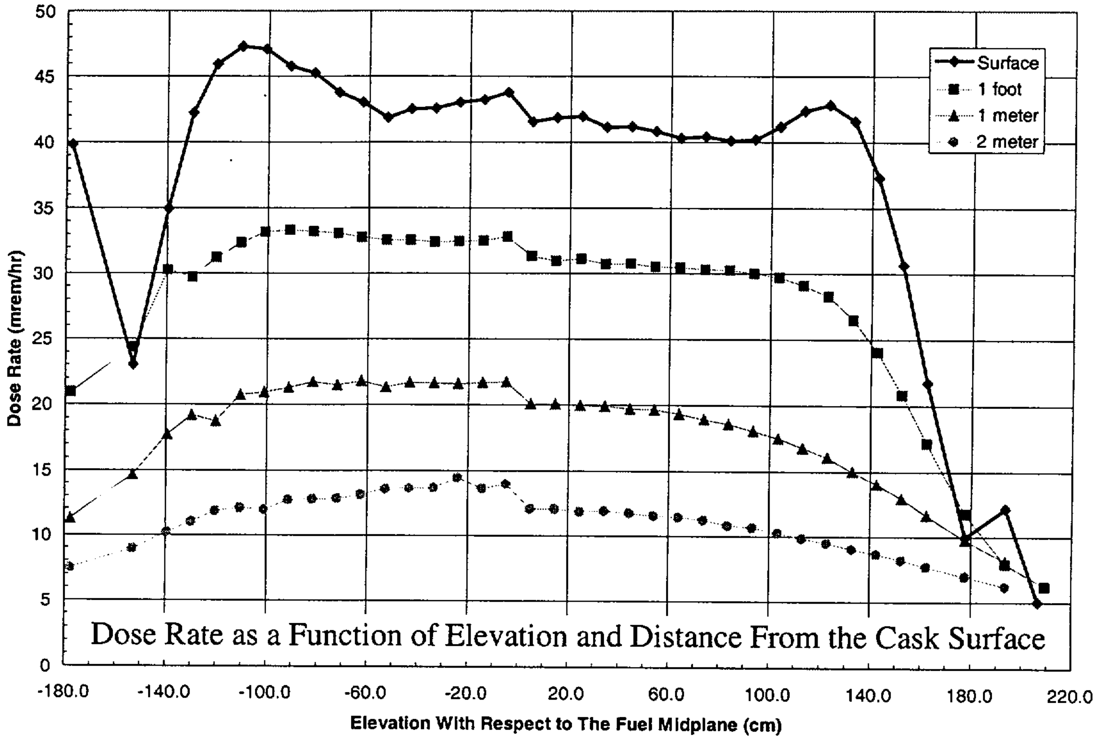


Figure 5.4-2 Storage Cask Three-Dimensional Model Surface Dose Rate Profile

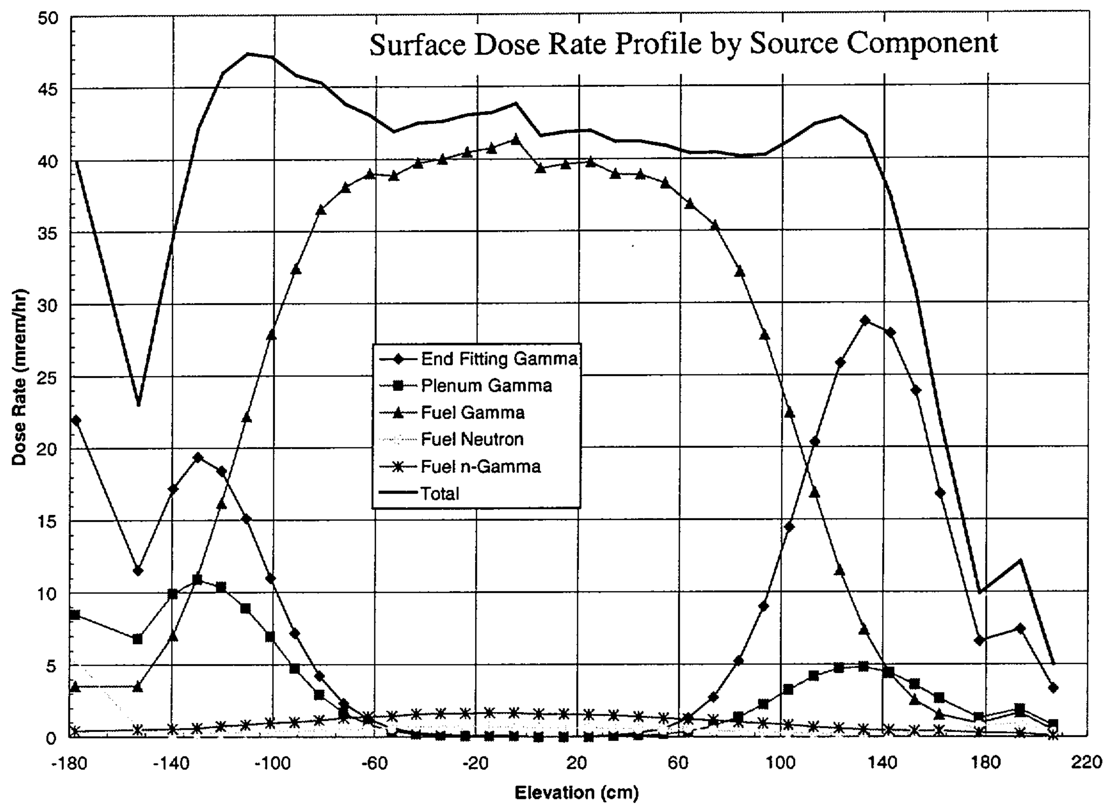


Figure 5.4-3 Storage Cask Top Outlet Vent Dose Rate Profile

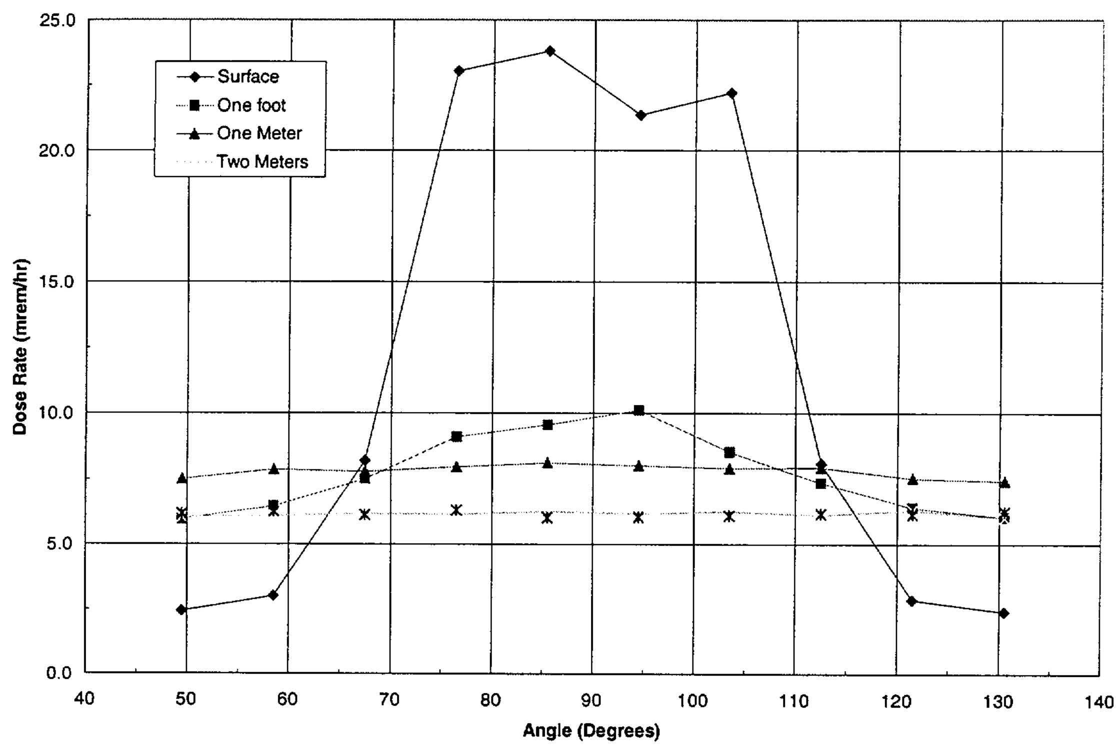


Figure 5.4-4 Storage Cask Bottom Inlet Vent Dose Rate Profile

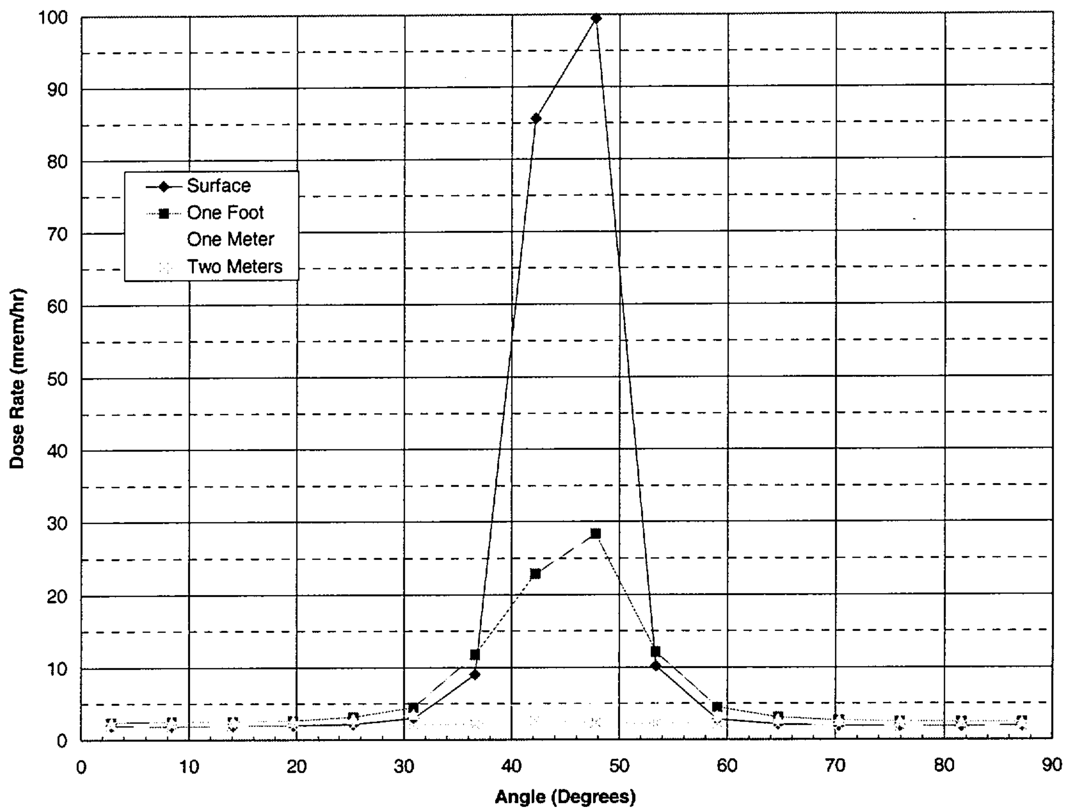


Figure 5.4-5 Storage Cask Dose Rate Profile at Stand Cutout Elevation

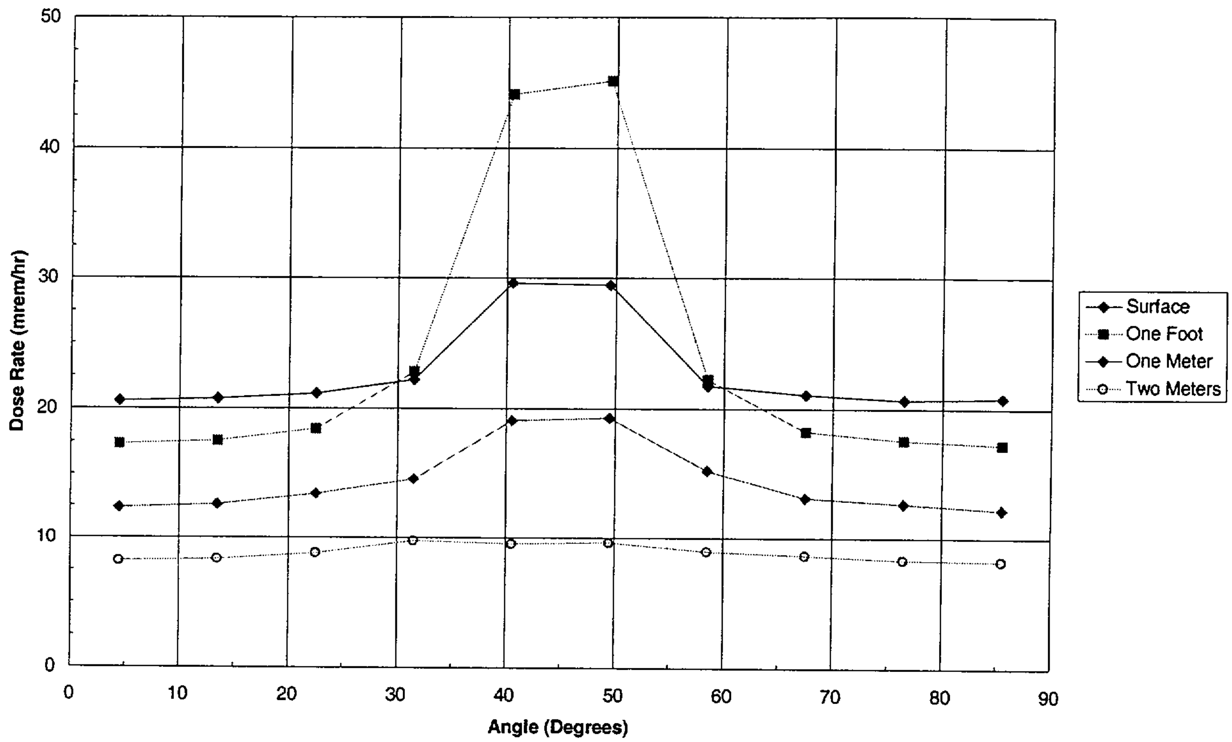


Figure 5.4-6 Storage Cask Top Dose Rate Profile as a Function of Radius from the Centerline and Distance from Surface

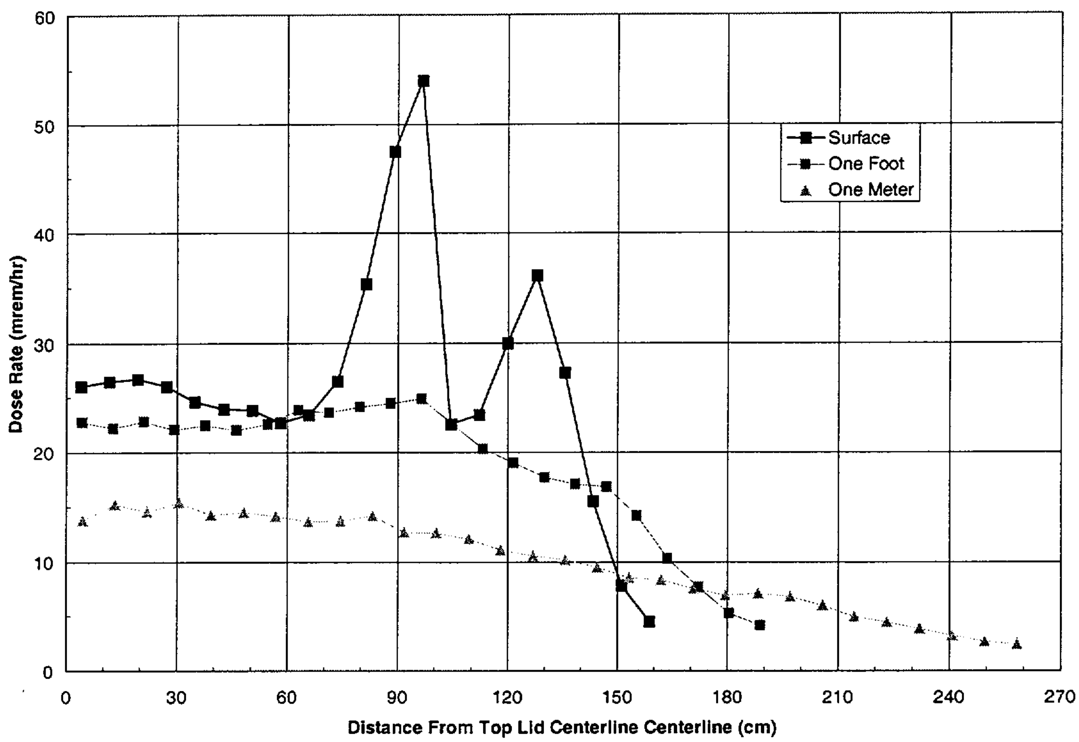


Figure 5.4-7 Storage Cask Top Surface Dose Rate Profile by Source Component

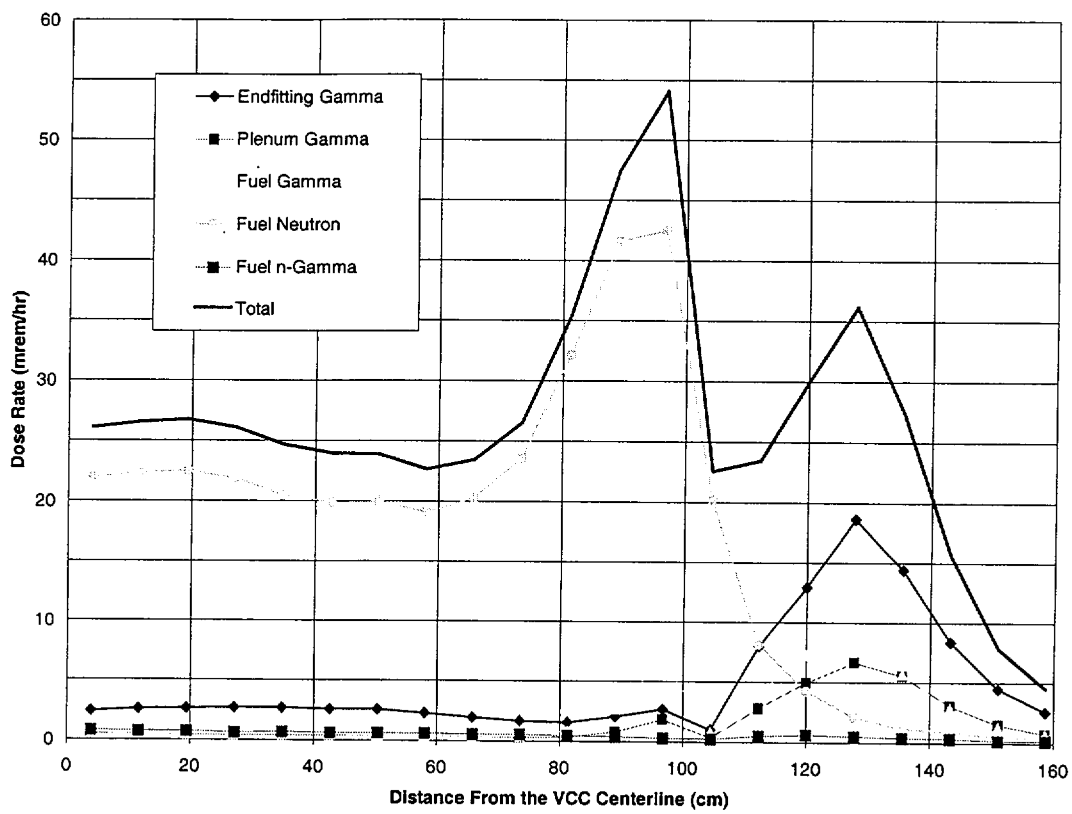


Figure 5.4-8 Transfer Cask Side Dose Rate Profile as a Function of Elevation and Distance, Wet Cavity

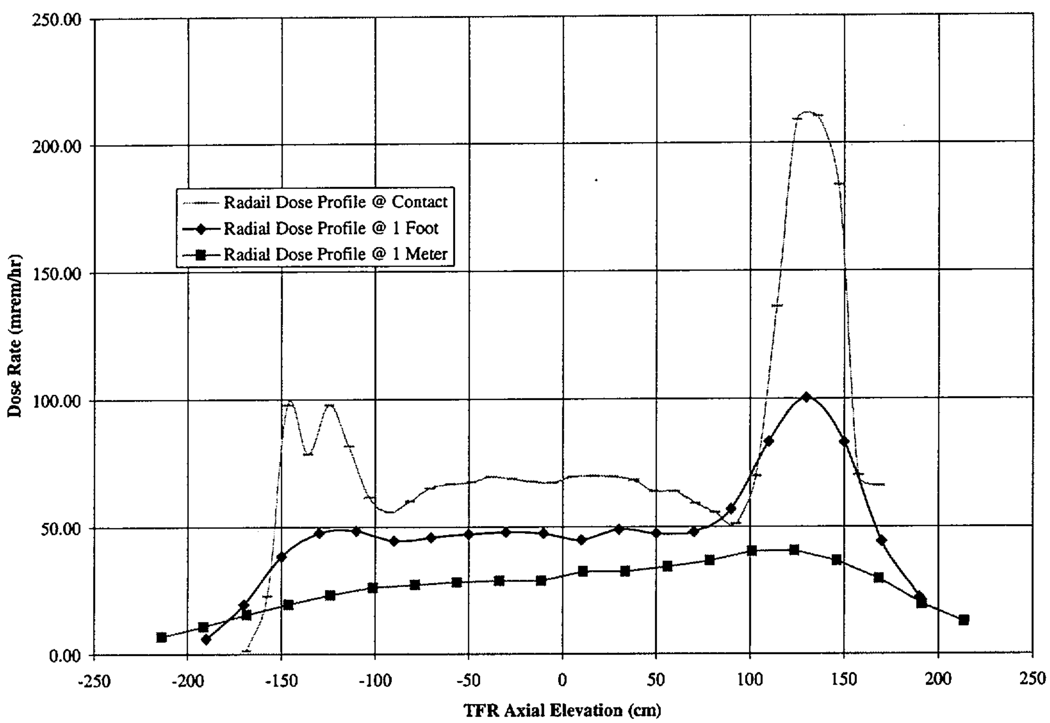
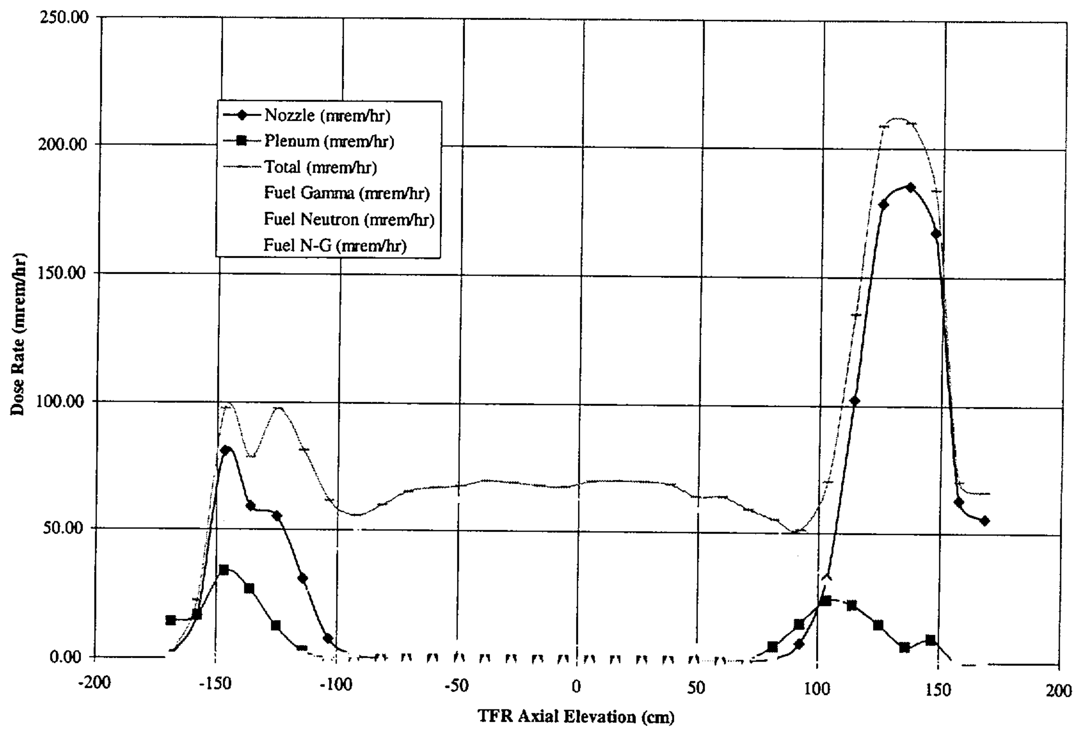


Figure 5.4-9 Transfer Cask Side Dose Rate Profile by Source Component, Wet Cavity



Note: Peak in nozzle dose at 135 cm is due to draining 50 gallons of water which uncovers the nozzle.

Figure 5.4-10 Transfer Cask Side Dose Rate Profile as a Function of Elevation and Distance, Dry Cavity

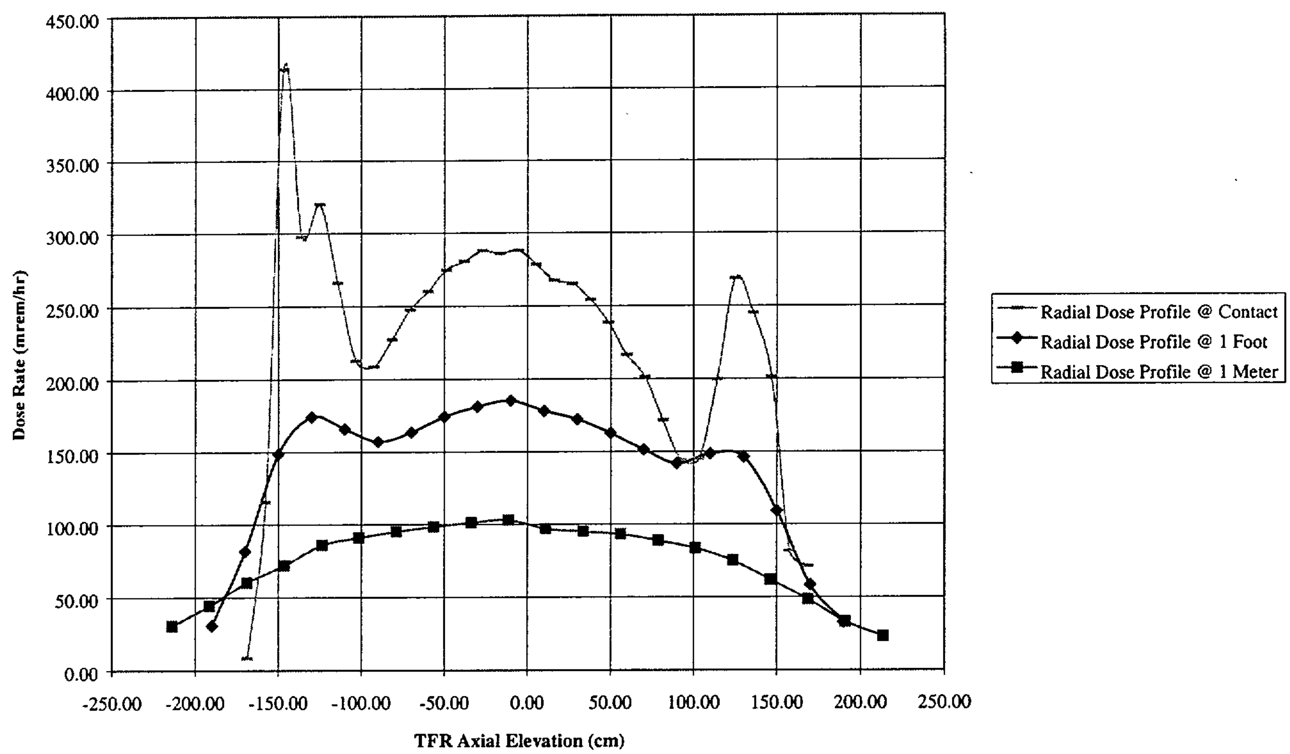


Figure 5.4-11 Transfer Cask Side Dose Rate Profile by Source Component, Dry Cavity

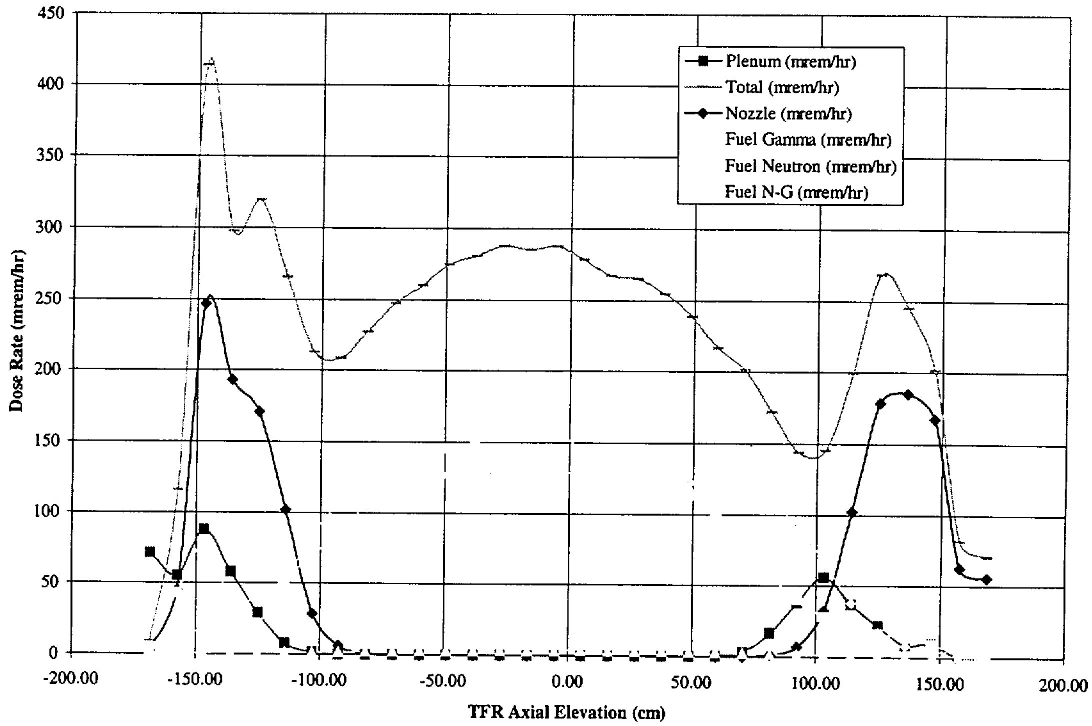


Figure 5.4-12 Transfer Cask Top Surface Dose Rate as a Function of Radius and Distance from Surface, Shield Lid, Structural Lid, and Temporary Shield On, Dry Cavity

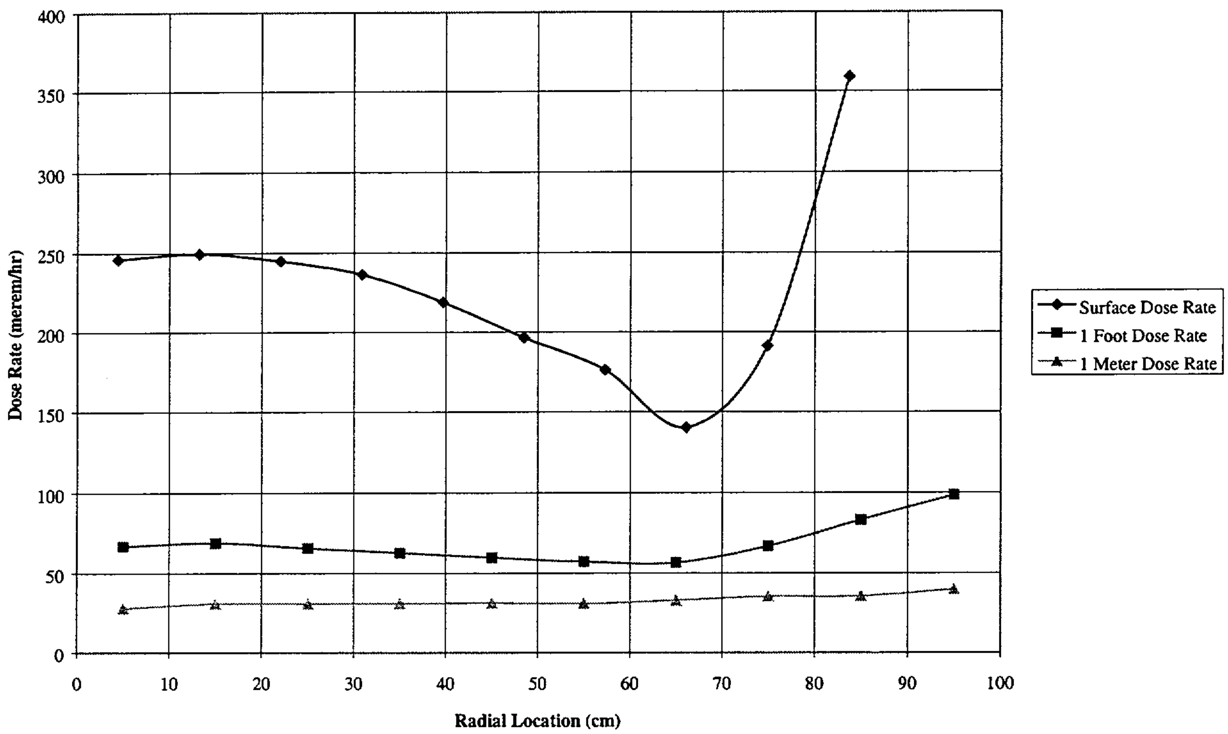


Figure 5.4-13 Transfer Cask Top Surface Dose Rate by Source Component, Shield Lid, Structural Lid, and Temporary Shield On, Dry Cavity

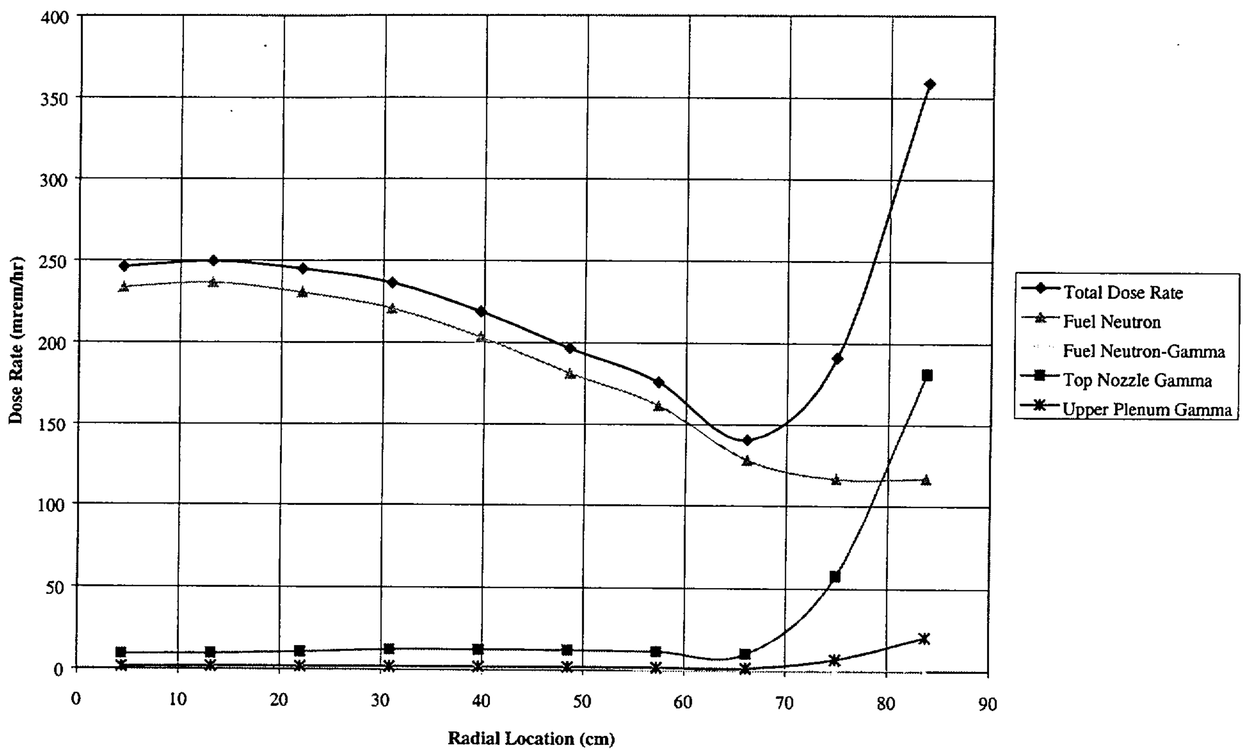


Figure 5.4-14 Transfer Cask Bottom Surface Dose Rate, Wet Cavity

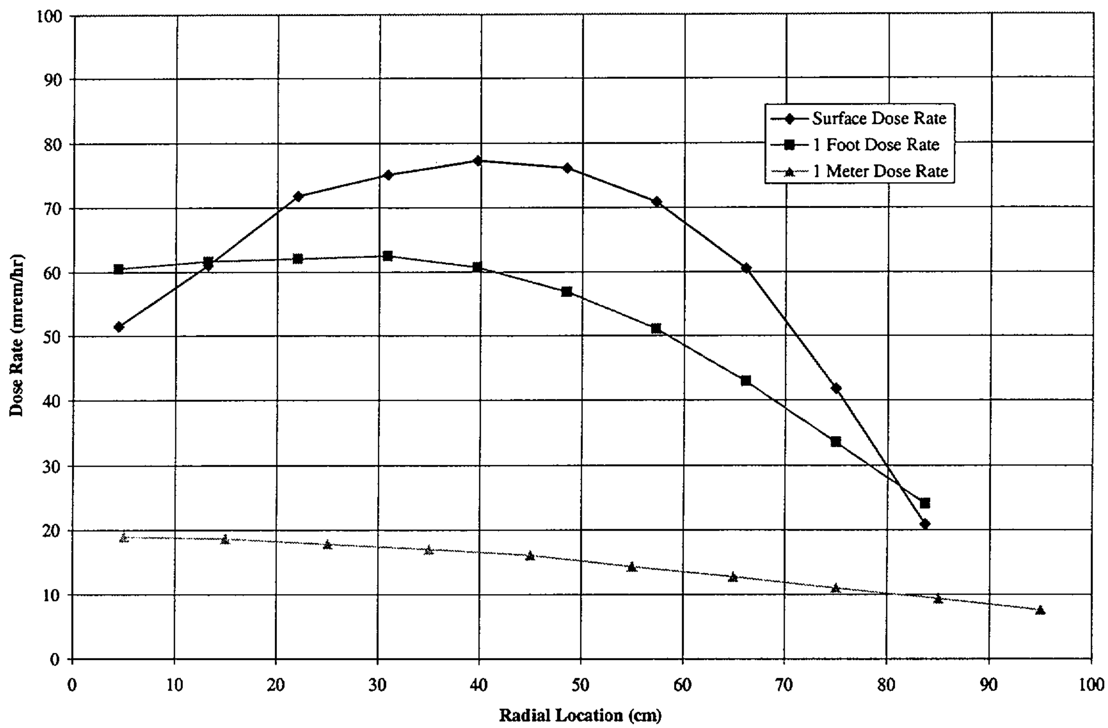


Figure 5.4-15 Transfer Cask Bottom Surface Dose Rate, Dry Cavity

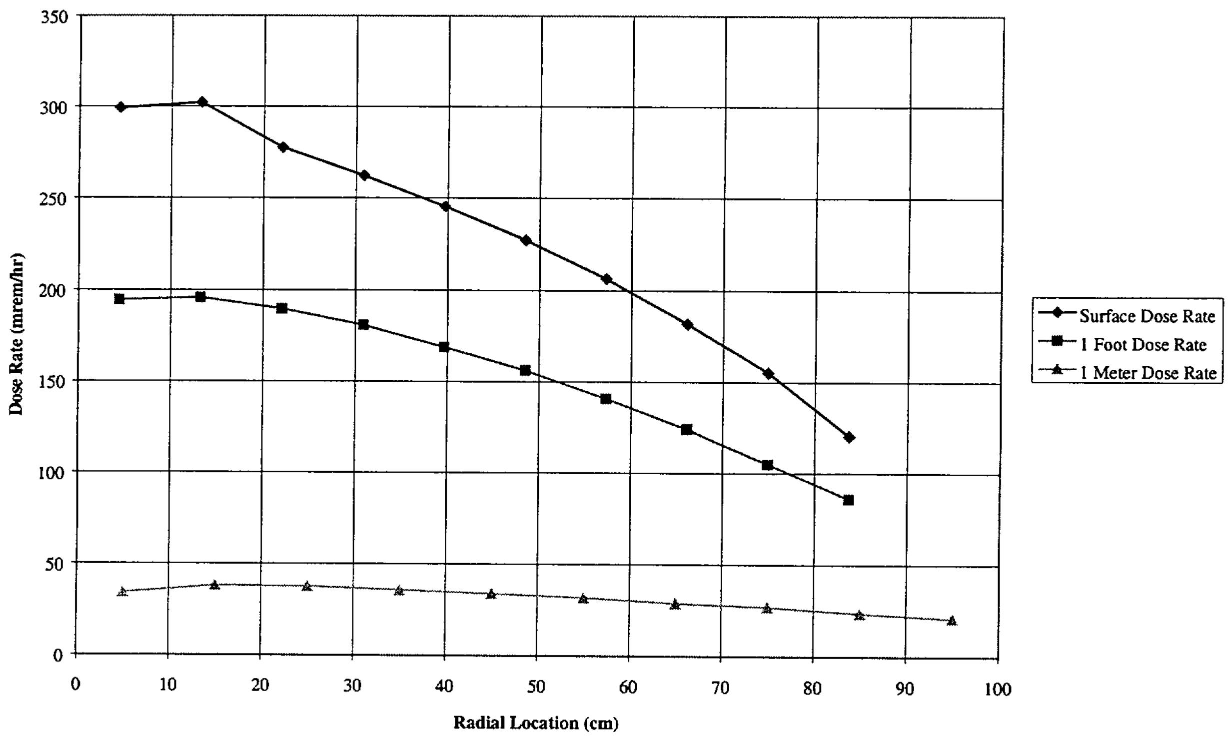


Table 5.4-1 ANSI Standard Neutron Flux-To-Dose Rate Factors

Group	(Rem/Hr)/(N/cm^2/Sec)
1	1.49160E-04
2	1.44640E-04
3	1.27010E-04
4	1.28110E-04
5	1.29770E-04
6	1.02810E-04
7	5.11830E-05
8	1.23189E-05
9	3.83650E-06
10	3.72469E-06
11	4.01500E-06
12	4.29259E-06
13	4.47439E-06
14	4.56760E-06
15	4.55809E-06
16	4.51850E-06
17	4.48790E-06
18	4.46649E-06
19	4.43450E-06
20	4.32709E-06
21	4.19750E-06
22	4.09759E-06
23	3.83900E-06
24	3.67480E-06
25	3.67480E-06
26	3.67480E-06
27	3.67480E-06

Table 5.4-2 ANSI Standard Gamma Flux-To-Dose Rate Factors

Group	(Rem/Hr)/(N/cm^2/Sec)
1	8.77160E-06
2	7.47849E-06
3	6.37479E-06
4	5.41360E-06
5	4.62209E-06
6	3.95960E-06
7	3.46860E-06
8	3.01920E-06
9	2.62759E-06
10	2.20510E-06
11	1.83260E-06
12	1.52280E-06
13	1.17250E-06
14	8.75940E-07
15	6.30610E-07
16	3.83380E-07
17	2.66930E-07
18	9.34720E-07

Table 5.4-3 NAC-MPC Storage Cask One-Dimensional Projectile Accident Radial Dose Rates

Source	Surface (mrem/hr)	1 meter (mrem/hr)
Fuel Neutron	4.1	1.5
Fuel Gamma	306.0	136.0
Fuel N-Gamma	4.5	1.8
Total	315	139

Table 5.4-4 NAC-MPC Shielded Gamma Flux

Energy Group Boundary (MeV)	**Radial Gamma-Ray Group Flux (γ/sec/cm^2)**	**Axial Hardware Gamma-Ray Group Flux (γ/sec/cm^2)**	**Axial Fuel Gamma-Ray Group Flux (γ/sec/cm^2)**
8.0-10.0	1.76E+00	0.00E+00	9.26E-01
6.5-8.0	1.36E+01	0.00E+00	8.63E+00
5.0-6.5	1.71E+01	0.00E+00	4.04E+00
4.0-5.0	1.98E+01	0.00E+00	3.09E+00
3.0-4.0	2.72E+01	4.52E-24	3.74E+00
2.5-3.0	1.77E+01	5.27E-05	2.59E+00
2.0-2.5	9.40E+01	1.73E-02	5.37E+00
1.66-2.0	8.15E+01	1.39E-02	4.90E+00
1.33-1.66	4.65E+02	8.23E+01	1.45E+01
1.0-1.33	1.17E+03	2.05E+02	3.48E+01
0.8-1.0	1.56E+03	1.81E+02	3.26E+01
0.6-0.8	3.10E+03	2.48E+02	4.89E+01
0.4-0.6	5.53E+03	3.42E+02	7.96E+01
0.3-0.4	4.01E+03	2.06E+02	4.83E+01
0.2-0.3	5.63E+03	2.18E+02	5.20E+01
0.1-0.2	1.38E+04	1.37E+02	3.19E+01
0.05-0.1	4.62E+03	3.48E+00	8.14E-01
0.01-0.05	1.75E+01	5.24E-03	5.39E-03
Total Group Flux	4.02E+04	1.62E+03	3.77E+02
Total Source Strength (γ/sec)	1.18E+10[1]	1.34E+08[2]	3.12E+07[2]

1. Total radial source is total flux multiplied by the radial cask area (2.905×10^5 cm^2).
2. Total axial source is total flux multiplied by cask axial surface area (8.301×10^4 cm^2).

Table 5.4-5 NAC-MPC Shielded Neutron Flux

Energy Group Boundary (MeV)	Total Radial Neutron Group Flux (n/sec/cm^2)	Total Axial Neutron Group Flux (n/sec/cm^2)
6.43 - 20.0	3.82E-02	4.25E-02
3.0 - 6.43	1.42E-01	1.79E-01
1.85 - 3.0	3.42E-01	6.21E-01
1.4 - 1.85	1.71E-01	8.13E-01
0.9 - 1.4	1.56E-01	5.60E+00
0.4 - 0.9	3.51E-01	3.94E+01
0.1 -0.4	3.98E-01	7.47E+01
1.7E-02 - 0.1	3.62E-01	5.15E+01
3.0E-03 - 1.7E-02	2.85E-01	2.78E+01
5.5E-04 - 3.0E-03	3.25E-01	1.44E+01
1.0E-04 - 5.5E-04	4.04E-01	1.46E+01
3.0E-05 -1.0E-04	3.23E-01	9.36E+00
1.0E-05 - 3.0E-05	3.43E-01	7.76E+00
3.05E-06 - 1.0E-05	4.09E-01	7.06E+00
1.77E-06 - 3.05E-06	2.08E-01	3.00E+00
1.3E-06 -1.77E-06	1.28E-01	1.55E+00
1.13E-06 - 1.3E-06	6.01E-02	6.65E-01
1.0E-06 - 1.13E-06	5.37E-02	5.49E-01
8.0E-07 - 1.0E-06	1.00E-01	9.59E-01
4.0E-07 - 8.0E-07	3.62E-01	2.69E+00
3.25E-07 - 4.0E-07	1.35E-01	6.64E-01
2.25E-07 - 3.25E-07	5.56E-01	1.00E+00
1.0E-07 - 2.25E-07	6.37E+00	1.67E+00
5.0E-08 - 1.0E-07	1.37E+01	1.05E+00
3.0E-08 - 5.0E-08	9.49E+00	4.35E-01
1.0E-08 - 3.0E-08	8.35E+00	2.26E-01
1.0E-10 - 1.0E-08	1.65E+00	3.11E-02
Total Group Flux	4.52E+01	2.68E+02
Total Source Strength (n /sec)	1.31E+07[1]	2.223E+07[2]

1. Total radial source is total flux multiplied by radial cask area of 2.905 x 10^5 cm^2.
2. Total axial source is total flux multiplied by cask axial surface area of 8.301 x 10^4 cm^2.

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